ml030690029.ocr

Duke Power
Catawha Nuckar Station
PDAPDuke
4800 Concord Road
ower~
~ ~ ~
York, SC 29745
G R Pn
(803)8314251 OFFiCE
wacFYep,,,;r;;~rso
(803) 831-3221 FAK

February 25, 2003

U.S. Nuclear Regulatory Coromission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: Duke Energy Corporation
Catawba Nuclear Station, Units 1 and 2
Docket Numbers 50-413 and 50-414
Proposed Technical Specifications (TS) Amendments
TS Bases B 3.4.4, RCS Loops - MODES 1 and 2
TS Bases B 3.4.5, RCS Loops - MODE 3
TS Bases B 3.4.6, RCS Loops - MODE 4
TS Bases B 3.4.7, RCS Loops - MODE 5, Loops Filled
TS 3.4.13 and TS Bases B 3.4.13, RCS Operational LEAKAGE
New TS 3.4.18 and new TS Bases B 3.4.18, Steam Generator Tube Integrity
TS 5.5.9, Steam Generator (SG) Tube Surveillance Program
TS 5.6.8, Steam Generator Tube Inspection Report
Revision to Steam Generator TS

Pursuant to 10 CFR 50.4 and 10 CFR 50.90, Duke Energy Corporation hereby requests amendments to the Operating Licenses and TS to incorporate the changes described herein for Catawba Units 1 and 2.

In July of 1993, the industry and NRC initiated discussion concerning TS requirements for steam generators. These amendment requests represent the culmination of this discussion and are based upon the industry initiative known as the NEI Generic License Change Package (GLCP). The proposed amendments are being submitted for Catawba on behalf of the industry to demonstrate the acceptability of the steam generator GLCP initiative developed through NEI 97-06, "Steam Generator Program Guidelines." The industry has been working closely with the NRC for the past decade to develop more appropriate steam generator TS. As a result of this initiative, steam generator safety and performance will be significantly improved. These amendment requests will

formalize the NRC review process and will demonstrate the acceptability of this performance based initiative for Catawba, as well as for the rest of the industry.

The proposed amendments add new TS 3.4.18 and Bases for Steam Generator Tube Integrity and revise the TS and Bases as indicated above.

These amendment requests provide a programmatic framework for monitoring and maintaining the integrity of steam generator tubes consistent with 10 CFR 50, Appendices A and B and Catawba's licensing basis. This framework includes performance criteria that, if satisfied, provide reasonable assurance that tube integrity is being maintained. In addition, this framework provides for monitoring and maintaining the tubes to provide reasonable assurance that the performance criteria are met at all times between scheduled inspections of the tubes.

Catawba's Steam Generator Program will meet the intent of the guidance provided in the Steam Generator Integrity Elements section of NEI 97-06, "Steam Generator Program Guidelines," as it may be revised from time to time. The basis for any deviations from the intent of NEI 97-06 or its referenced EPRI guideline documents will be documented internally as part of the program implementation. Catawba's approach to the content and maintenance of its Steam Generator Program as described above will be established as a commitment in Catawba's commitment tracking system.

The changes to TS 3.4.13 reference Catawba's Steam Generator Program described in TS 5.5.9 for the Surveillance Requirements necessary to verify primary to secondary leakage. The proposed amendments also delete the existing Limiting Condition for Operation (LCO) 3.4.13.d. and revise the Conditions and Surveillance Requirements to clarify the requirements related to primary to secondary leakage.

New TS 3.4.18 and its Bases describe the approved steam generator performance criteria and establish actions that are necessary should the performance criteria not be met. Licensee initiated changes to the TS 3.4.18 Bases will be controlled by the TS Bases Control Program under the provisions of 10 CFR 50.59.

The changes to TS 5.5.9 require the implementation of a Steam Generator Program, describe several key elements of the program, and list the performance criteria, repair

criteria, repair methods, and inspection intervals proposed for use at Catawba.

The changes to TS 5.6.8 define the requirement for, and the contents of, the steam generator tube inspection report. The existing requirement for a 12-month report is changed to a 120-day report, submitted only if the number of tubes exceeding the repair criteria during scheduled inservice inspections exceeds 1% of those inspected.

Finally, editorial changes are made to the Bases for TS 3.4.4, 3.4.5, 3.4.6, and 3.4.7 to reflect changes in nomenclature for TS 5.5.9.

In summary, the TS and Bases changes included herein remove the detailed inspection requirements from the TS and replace them with the essential elements of a Steam Generator Program that includes significant enhancements to the existing TS. These proposed revisions will enhance the safety function of the steam generators by increasing the probability that the integrity of the steam generator tubes will be maintained between scheduled inservice inspections.

The contents of this amendment request package are as follows:

Attachment 1 provides a marked copy of the affected TS and Bases pages for Catawba, showing the proposed changes. Included in Attachment 1 is the proposed version of TS 3.4.18 and Bases. Attachment 2, containing the reprinted pages of the affected TS and Bases pages, will be provided to the NRC upon issuance of the approved amendments. Attachment 3 provides a background, description of the proposed changes, and technical justification. Pursuant to 10 CFR 50.92, Attachment 4 documents the determination that the amendments contain No Significant Hazards Considerations. Pursuant to 10 CFR 51.22(c)(9), Attachment 5 provides the basis for the categorical exclusion from performing an Environmental Assessment/Impact Statement.

Implementation of these amendments to the Catawba Facility Operating Licenses and TS will impact the Catawba Updated Final Safety Analysis Report (UFSAR). The affected UFSAR section is pending UFSAR Chapter 18, "Aging Management Programs and Activities." Necessary UFSAR changes will be submitted in accordance with 10 CFR 50.71(e).

The NRC has had considerable input into the development of the enclosed TS changes and has been actively involved in the discussion of the relevant technical issues. Therefore, Duke Energy Corporation is requesting NRC review and approval of these proposed amendments by June 30, 2003 in order to allow implementation on a timely basis and to allow other licensees to submit similar amendment requests. Duke

Energy Corporation is requesting a 60-day implementation period in conjunction with these amendments. In addition, pursuant to 10 CFR 170.11 (a) (1) (iii), since these proposed amendments are being submitted on behalf of the industry as a lead plant submittal, Duke Energy Corporation is requesting an exemption from licensing fees associated with the review and approval of these requests. Consistent with the cited regulation, these amendment requests represent a means of exchanging information between industry organizations and the NRC for the specific purpose of supporting the NRC's generic regulatory improvements or efforts.

In accordance with Duke Energy Corporation administrative procedures and the Quality Assurance Program Topical Report, these proposed amendments have been previously reviewed and approved by the Catawba Plant Operations Review Committee and the corporate Nuclear Safety Review Board.

Pursuant to 10 CFR 50.91, a copy of these proposed amendments is being sent to the appropriate State of South Carolina official.

Inquiries on this matter should be directed to L.J. Rudy at (803) 831-3084.

Very tru

Gary R. Peterson

URIs

Attachments

D~cument Control Desk
Page 5
February 25, 2003

Gary R. Peterson affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of

~¼ ~~snowled~e.
~Gary R Peterson, Vice President
Subscribed and sworn to me: -33
Date

My commission expires:
Date

--

SEAL
Document Control Desk
Page 6
February 25, 2003

xc (with attachments):

L.A. Reyes
U.S. Nuclear Regulatory Commission
Regional Administrator, Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30303

E.F. Guthrie
Senior Resident Inspector (CNS)
U.S. Nuclear Regulatory Commission
Catawba Nuclear Station

R.E. Martin (addressee only)
NRC Senior Project Manager (CNS)
U.S. Nuclear Regulatory Commission
Mail Stop 08-H12
Washington, D.C. 20555-0001

H.J. Porter, Director
Division of Radioactive Waste Management
Bureau of Land and Waste Management
Department of Health and Environmental Control
2600 Bull St.
Columbia, SC 29201

ATTACHMENT 1

MARKED-UP TS AND BASES PAGES FOR CATAWBA

TABLE OF CONTENTS (continued)

p~J

TABLEOFCONTENTS ~

BASES

APPLICABLE SAFETY ANALYSES (continued)

assuming the number of RCS loops in operation is consistent Wiw1 the Technical Specifications. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The primary coolant flowrate, and thus the number of RCPs in operation, is an important assumption in all accident analyses (Ref. 1).

Steady state DNB analysis has been performed for the four RCS loop operation. For four RCS loop operation, the steady state DNB analysis, which generates the pressure and temperature Safety Limit (SL) (i.e., the departure from nucleate boiling ratio (DNBR) limit) assumes a maximum power level of 118% RTP. This is the design overpower condition for four RCS loop operation. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.

The plant is designed to operate with all RCS loops in operation to maintain DNBR above the SL, during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant.

RCS Loops-MODES 1 and 2 satisfy Criterion 2 of 10 CFR 50.36 (Ref. 2).

LCO The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required in MODES 1 and 2.

An OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an OPERABLE SG in accordance with the Steam Ge tor

APPLICABILITY In MODES 1 and 2, the reactor is critical and thus has the potentia l to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

The decay heat producUon rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3,4, and 5.

Operation in other MODES is covered by:

BASES

LCO (continued)

Utilization of the Note is permitted provided the following conditions are met, along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration, thereby maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 1 00F below saturation temperature1 so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

An OPERABLE RCS loop consists of one OPERABLE and one OPERABLE SG in accordance with the Steam Generator~~((~)~ (~IlceProgram, which has the minimum water level specified in SR 3A.5.2. An ROP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY In MODE 3, this LCO ensures forced circulation of the reactor coo lant to remove decay heat from the core and to provide proper boron mixing. The most stringent condition of the LCO, that is, three RCS loops OPERABLE and three RCS loops in operation, applies to MODE 3 with RTBs in the closed position. The least stringent condition, that is, three RCS loops OPERABLE and one RCS loop in operauon, applies to MODE 3 with the RTBs open.

Operation in other MODES is covered by:

LCO 3A.4, "RCS Loops-I~ODES 1 and 2";
LCO 3A.6, "RCS Loops~MODE 4";
LCO 3.4.7, "RCS Loops~MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled";
LCO 3.4.17, "RCS Loops-Test Exceptions";
LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulafior-High Water Level" (MODE 6); and

LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level~ (MODE 6).

BASES

LCO (continued)

performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits the de-energizing of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characterisUcs of the RCS, the input values must be revalidated by conducting the test again. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow
.

Utilization of Note 1 is permitted provided the following conditions are me
t
along with any other condiUons imposed by initial startup test procedure
s:

a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 1 00F below saturaUon temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 requires that the secondary side water temperature of each SG be <500F above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature < 2850F. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

An OPERABLE RCS loop comprises an OPERABLE RCP~an(;~~~ OPERABLE SG in accordance with the Steam Generator~ ~llcProgram, which has the minimum water level specified in SR 3.4.6.2. The water level is maintained by an OPERABLE AFW train in accordance with LCO 3.7.5, "Auxiliary Feedwater System."

Similarly for the RHR System, an OPERABLE RHR loop comprises an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RCPs and RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

LCO (continued)

This restriction is to prevent a low ',9mperature overpressure event due to

a thermal transient when an RCP is started.

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops.

An OPERABLE RHR loop is comprised of an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. If not in its normal RHR alignment from the RCS hot leg and returning to the RCS cold legs, the required RHR loop is OPERABLE provided the system may be placed in service from the control room, or may be placed in service in a short period of time by actions outside the control room an

d

there are no restraints to placing the equipment in service. RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. An OPERABLE SG can perform as a heat sink when it has an adequat and is OPERABLE in accordance with the Steam era

APPLICABILITY In MODE 5 with RCS loops filled, this LCO requires forced circulation of

the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side narrow range water level of at least two SGs is required to be >- 12%.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops-MODES 1 and 2":
LCO 3.4.5, "RCS Loops-MODE 3";
LCO 3.4.6, I•RCS Loops~MODE 4";
LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled";
LCO 3.4.17 "RCS Loops-Test Exceptions";
LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation-High Water Level" (MODE 6); and
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level" (MODE 6).

ACTIONS A.1 and A.2

If one RHR loop is inoperable and the required SGs have secondary side

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;
b. 1 gpm unidentified LEAKAGE;
C. 10 gpm identified LEAKAGE;

(d. 57 gallons per day tal primary to se$ndary LEAKA~ through all~ earn generators (IGS); and I

¾ 150 gallons per day primary to secondary LEAKAGE through any one~

~ ~ (~

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
ROS LEAKAGE not within limits for reasons (~(?('~r~~) other than pressure boundary LEAKAG	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

~~t~atawbaUnits1 and 2 3.4.13-1 Amendment Nos.~5

RCS Operational LEAKAGE

3.4.13

SURVEILLANCE REQUIREMENTS (½

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	------- -------NOT~ -~---------- ~ Not required to be performed~4ntil 12 hoursiof steady state operation. ------------ ---------------~--------- ------- ~ ~ ~ P~~ry Verify RCS Operational LEAKAGE within limits by performance of RCS water inventory balance.	Only required to be performed during steady state operation 72 hours
SR 3.4.13.?	flenfy~am gener or tube integ/ty is in dance with t Steam Ge erator Tube $urveillance ~rogram.) Y~rift pri~~r~ ~ 4 eli ce	In accordance with the Steam Generator U e

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operauonal LEAKAGE

BASES

BACKGROUND Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolung, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operaUon in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detecUon systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecUng and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the idenUfied LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradati6n arid the core from inadequate cooling, in addition to preventing the a~ccident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

APPLICABLE Except for primary to secondary LEAKAGE, the safety analyses d
o not
SAFETY ANALYSES address operational LEAKAGE. However, other operaUonal L
EAKAGE is
related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event.

The safety analysis (Ref. 3) for an event resultin in steam dis e to

reactor coolant system WI continue to leak water inventory to the secondary side, and in which there will be a postulated source term associated with the accident, uUlizes this leakage value as an input in the analysis. These accidents include the rod ejection accident, locked rotor accident, main steam line break, steam generator tube mpture and uncontrolled rod withdrawal accident. The rod election accident, locked rotor accident and uncontrolled rod withdrawal accident yield a source term due to postulated fuel failure as a result of the accident. The main steam line break and the steam generator tube rupture yield a source ter
m
due to perforations in fuel pins causing an iodine spike. Primary to secondary side leakage may escape the secondary side due to flashing or atomization of the coolant, or it may mix with the secondary side SG wate
r
inventory and be released due to steaming of the SGs. The rod ejection accident is limiUng compared to the remainder of the accidents with respect to dose results. The dose results for each of the accidents delineated above are well within 10 CFR 100 limits for the rod ejection accident, and below a small fracUon of 10 CFR 100 limits for the remainder of the accidents.

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36 ½

~

(Ref. 4).

LCO RCS operational LEAKAGE shall be limited to:

a. Pressure Boundarv LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicaUve of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE.

Violation of this LC'd co'u~ld result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment

Catawba Units 1 and 2 B 3.4.13-2 Revision No.~

INSERT A for B 3.4.13 Applicable Safety Analyses:

that primary to secondary LEAKAGE from each steam generator (SG) is 150 gallon
s per
day

RCS Operational LEAKAGE

B 3.4.13

BASES

LCO (continued)

can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

C. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified or total LEAKAGE and is well within the capability of the ROS Makeup System. Identified LEAKAGE includes LEAKAGE captured by the pressurizer relief tank and reactor coolant drain tank, as well as quantified LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Pri a to Seconda AKAGE throu h A Steam Generat 5
5

otal primary to sec ndary LEAKAGE a ounting to 576 g through all SGs pr duces acceptable 0 site doses in the ccident analysis. ViolaUo of this LCO could ceed the offsite d se limits for the previousl described accident . Primary to seco ary LEAKAGE mu be included in the t tal allowable limit f r identified LEAKAGE.

¾ Primary to Secondary LEAKAGE through Any One SG

APPLICABILITY In MODES 1, 2, 3, and 4,~t~epotential for RCPB LEAKAGE is great
est
when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

Catawba Units 1 and 2 B 3.4.13-3 Revision No.~ ~

INSERT B for B 3.4.13 LCO:

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in the Steam Generator Program. Steam Generator Program requirements are governed by NEI 97-06, "Steam Generator Program Guidelines~ (
Ref.
6). The Steam Generator Program operational LEAKAGE performance criterion sta
tes:
"The RCS operational primary to secondary LEAKAGE through any one SG shall b
e
limited to 150 gallons per day."

The primary to secondary LEAKAGE measurement is based on the methodology described in Ref. 5. Currently, a correction factor is applied to account for the fact that
current safety analyses take the primary to secondary leak rate at reactor coolant conditions, rather than at room temperature as described in Ref. 5.

The operational LEAKAGE rate limit applies to LEAKAGE in any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the LEAKAGE should be conservatively assumed to be from one SG.

The limit in this criterion is based on operating experience gained from SG tube degradation mechanisms that result in tube LEAKAGE. The LEAKAGE rate criterio
n in
conjunction with the implementation of the Steam Generator Program provides reasonable assurance that a single flaw leaking this amount will not propagate to
a SG
tube rupture under normal and accident conditions prior to detection by LEAKAGE monitoring methods and commencement of plant shutdown. If LEAKAGE is throug
h
more than one flaw, the flaws are smaller than the assumed limiting flaw and the a
bove
assumption is conservative.

APPLICABILITY (continued)

LCO 3.4.14, e•RCS Pressure Isolation Valve (PIV) Leakag~1u measures leakage through each individual PIV and can impact this LCO. ~f the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable unidentified LEAKAGE.

ACTIONS A.1

Unidentified LEAKAG E~dentified LEAKAG E~~~aryosecoa
~in excess of the LCO limits must be reduced to within limits
within 4 hours. This Completion Time allows time to verify leakage rates

and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.
of;~ ~ ~
Bland B.2 ~

If any pressure bound~ LEAKAGE exists~or if unidentified LEAKAG~~ identified LEAKAG or rima second LEA GE annot be reduced to within limits within 4 ho~tereactormustbe brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. I
n
MODE 5, the pressure stressesacung on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE SR 3.4.13.1
REQUIREMENTS
Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified

SURVEILLANCE REQUIREMENTS (continued)

LEAKAGE are determined b erformance of an RCS water inventory balance. Primary to sec ndary LEAKAG is also measure y' pe or ance of an RC water inventory lance in conjunc on with efflue t monitoring wit in the secondary team and feedw er Systems. For this SR, the volumetric calculation of unidentified LEAKAGE and identified LEAKAGE is based on a densi at room tem erature of 77 de rees F. The volu tric calculation primary 10 seco dary LEAKAGE is base on a density t operating RC temperature of 5 degrees F.

In order to rovide enhanced ssurance that the pr ary to second ry LEAKAG limit of LCO 3.4.1 is_met in MOpE 1, continuous cal ulation is perfor edv7~~a:nOperato Aid Computer progr m that utilizes e raUo of prim and secondary 5 tern activities to del rmine a LEA GE rate.
________ This ve ification methodolo y is based on guida ce contained i Re
f. 5.

In add ion, on a monthly sis, primary to seco dary LEAKAG is deter med based on gra samples.

__________________ The RCS water inventory balance must be performed with t
he reactor at
steady state operating conditions and near operatin~ure.
~~efore,this SR is not required lobe complete ES 3 d until
12 hours of steady state operation near operating pressure ave een established.

Steady state operation is required to perform a proper inventory bala
nce;
calculations during maneuvering are not useful and~Note~irequires t
he Qi
Surveillance to be met when steady state is established. For RCS operational LEAKAGE determination by water inventory balance, stea
dy
state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP s
eal
injection and return flows.

~~1~An early warning of pressure boundary LEAKAGE or unidentified
LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump leve
l. It
should be noted that LEAKAGE past seals and gaskets is not pressur
e
boundary LEAKAGE. These leakage detection systems are specified i
n
LCO 3.4.15, "RCS Leakage Detection Instrumentation."

The 72 hour Frequency is a reasonable interval to trend LEAKAGE an
d

~reco nizes the importance of early~leakage detection in the preventi
on of
accidents. A Note under the Frequen~cy column states that this SR is
of ~ required to be performed during steady state operation.
Catawba Units 1 and 2 B 3A.13-5 Revision No.~ 3
INSERT C for B 3.4.13 Surveillance Requirements:

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE bec
ause
LEAKAGE of 150 gallons per day or lower cannot be measured accurately by an R
CS
water inventory balance.
RCS Operational LEAKAGE

B 3A.13
BASES C);

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.13.2

Surv illance cannot be p ormed at norrnal op rating conditi 5.
REFERENCES 1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. UFSAR, Section 15.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

5. EPRI TR-104788-R2, "PWR Pnmary-to-Secondary Leak
Guidelines," Revision 2.

(~, ~gj q7~o~, ~ ~ ~ ~ id) ½~)
Catawba Units 1 and 2 B 3.4.13-6 Revision No~ ?~
INSERT D for B 3.4.13 Surveillance Requirements:

This SR verifies that primary to secondary LEAKAGE is less than or equal to 150 g
allons
per day through any one SG. Satisfying the primary to secondary LEAKAGE limit
ensures that the operational LEAKAGE performance criterion in the Steam Generat
or
Program is met. If this SR is not met, complian6e with LCO 3.4.18 should be evalua
ted.
The 150 gallons per day limit is based on room temperature measurements.

The Surveillance Frequency is in accordance with the Steam Generator Program
requirements. The Steam Generator Program's primary to secondary LEAKAGE te
st
Frequency is based upon guidance provided in Ref. 5. During normal operation the
primary to secondary LEAKAGE is determined using continuous process radiation
monitors or radiochemical grab sampling. The Steam Generator Program may req
uire
the Frequency of monitoring and sampling to increase as the amount of detected
LEAKAGE increases or if there are no continuous process radiation monitors avail
able.

(A/~~~s ~ SG Tube Integrity
3.4.18

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.18 Steam Generator (SG) Tube Integrity

LCO 3.4.18 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube repair criteria shall be plugged or

repaired in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

NOTE

Separate Condition entry is allowed for each SG tube.

CONDITION REQUIRED ACTION COMPLETION TIME
A. One or more SG tubes A.1 Verify tube integrity of 7 days
satisfying the tube the affected tube(s) is
repair criteria and not maintained.
plugged or repaired in
accordance with the AND
Steam Generator
Program. A.2 Plug or repair the Prior to entering
affected tube(s) in MODE 4 following
accordance with the the next refueling
Steam Generator outage or SG
Program. inspection
B. Required Action and B.1 Be in MODE 3. 6 hours
associated Completion
Time of Condition A AND
not met.
B.2Be in MODE 5. 36 hours
OR

SG tube integrity not
maintained.
Catawba Units 1 and 2 3.4.18-1 Amendment Nos.
SG Tube Integrity
3.4.18
SURVEILLANCE REQUIREMENTS
SURVEILLANCE FREQUENCY
SR 3.4.18.1 Verify SG tube integrity is maintained in In accordance
accordance with the Steam Generator with the Steam
Program. Generator
Program
SR 3.4.18.2 Verify that each inspected SG tube that Prior to entering
satisfies the tube repair criteria is plugged or MODE 4
repaired in accordance with the Steam following a SG
Generator Program. inspection
Catawba Units 1 and 2 3.4.18-2 Amendment Nos.
SG Tube Integrity
B 3.4.18

B 3.4 REACTOR COOLANT SYSTEM (RCS)
B 3.4.18 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND SG tubes are small diameter, thin walled tubes that carry primar
y
coolant through the primary to secondary heat exchangers in pressurized water reactors (PWRs). In the context of this Specification, tubing is defined as:

"Steam generator tubing refers to the entire length of the tube, including the tube wall and any repairs made to it, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube."

The SG tubes have a number of important safety functions. SG tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied upon to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.4, "RCS Loops - MODES 1 and 2," LCO 3.4.5, "RCS Loops - MODE 3," LCO 3.4.6, "RCS Loops - MODE 4," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled."

Concerns relating to the integrity of SG tubing stem from the fact that the tubing is subject to a variety of degradation mechanisms. Throughout the industry, SG tubes have experienced degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. A means of determining and managing degradation is needed. SG performance criteria were developed for this purpose.

The SG performance criteria identify the standards against which performance is to be measured. Meeting the performance criteria

'I
BASES

BACKGROUND (continued)

provides reasonable assurance that the SG tubing remains capable of fulfilling its specific safety function of maintaining RCPB integrity. The SG performance criteria and the processes required to meet them are defined by the NEI Steam Generator Program Guidelines (Ref. 1).

There are three SG performance criteria: accident induced leakage, structural integrity, and operational LEAKAGE. They act together to provide reasonable assurance of tube integrity at normal and accident conditions. SG tube integrity means that the tubes are capable of performing their intended safety functions consistent with their licensing basis, including applicable regulatory requirements.

The purpose of this LCO is to require compliance with the SG performance criteria. The accident induced leakage and structural integrity performance criteria apply to SG tubes and associated appurtenances considered part of the SG primary to secondary pressure boundary (e.g., plugs, sleeves, and other repairs). The accident induced leakage and structural integrity performance criteria are documented in Specification 5.5.9.

The third performance criterion, operational LEAKAGE, is addressed by LCO 3.4.13, "RCS Operational LEAKAGE."

APPLICABLE SAFETY ANALYSES

Satisfying the SG structural integrity performance criterion provides reasonable assurance against tube burst and the resulting primary to secondary LEAKAGE that might occur at normal and accident conditions.

Satisfying the accident induced leakage performance criterion provides reasonable assurance of acceptable primary to secondary LEAKAGE that might occur as a result of design basis accident conditions other than a SG tube rupture. The consequences of design basis accidents that include primary to secondary LEAKAGE depend, in part, on the accident induced leakage and the radioactive source term in the primary coolant.

The design basis accidents for which the primary to secondary LEAKAGE is a pathway for release of activity to the environment include the main steam line break, SG tube rupture, reactor coolant pump locked rotor accident, single rod withdrawal accident, and rod ejection accident. The analysis of radiological consequences of these design basis accidents, except for a SG tube rupture, assumes that the total primary to secondary

BASES

APPLICABLE SAFETY ANALYSES (continued)

LEAKAGE from each SG initially is 150 gallons per day. Transient thermal hydraulic analyses of these design basis accidents determine the primary to secondary LEAKAGE changes (decreases or increases) that result from changing pressures and temperatures. These calculated values are used in the analyses of radiological consequences of these design basis accidents.

The source term in the primary coolant for some design basis accidents (e.g., reactor coolant pump locked rotor accident and rod ejection accident) is associated primarily with fuel rods calculated to be breached. For other design basis accidents (e.g., main steam line break and SG tube rupture), the source term in the primary coolant consists primarily of the levels of Dose Equivalent 1131 radioactivity levels calculated for the design basis accident. This, in turn, is based on the limiting values in the Technical Specifications and postulated iodine spikes.

For accidents in which the source term in the primary coolant consists of the Dose Equivalent 1131 activity levels, the SG tube rupture yields the limiting values for radiation doses at offsite locations. In the calculation of radiation doses following this event, the rate of primary to secondary LEAKAGE in the intact SGs is set equal to the operational LEAKAGE rate limits in LCO 3.4.13. For the ruptured SG, a double ended rupture of a single tube is assumed. Following the initiating event, contaminants in flashed and atomized break flow (the latter computed for time spans during which the tubes are calculated to be uncovered), as well as secondary coolant, may be released to the atmosphere. Before reactor trip, the accident analysis for the SG tube rupture assumes that these contaminants are released to the condenser and from there to the environment with credit taken for scrubbing of iodine contaminants in the condenser. Following reactor trip (and loss of offsite power), the accident analysis assumes that these contaminants are released to the environment through the SG power operated relief valves and the main steam code safety valves until such time as the closure of these valves can be credited.

For other design basis accidents such as main steam line break, rod ejection accident, reactor coolant pump locked rotor accident, and uncontrolled rod withdrawal accident, the tubes are assumed to retain their structural integrity (i.e., they are assumed not to rupture). The LEAKAGE is assumed to be initially at the limit given in LCO 3.4.13. This is consistent with the accident induced leakage performance criterion.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The three SG performance criteria and the limits included in the plant Technical Specifications for Dose Equivalent 1131 in primary coolant and secondary coolant ensure the plant is operated within its analyzed condition. The dose consequences resulting from the most limiting design basis accident are within the limits defined in GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3), or the NRC approved licensing basis (e.g., a small fraction of these limits or 10 CFR 50.67 (Ref. 4)).

SG Tube Integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged or repaired in accordance with the Steam Generator Program.

During a SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is repaired or removed from service by plugging. If a tube was determined to satisfy the repair threshold but was not plugged or repaired, the tube may still have tube integrity.

SG tube integrity is defined by the performance criteria. The performance criteria include design basis parameters that define acceptable SG performance. The Steam Generator Program provides the evaluation process for determining conformance with the performance criteria.

Compliance with the LCO during MODES 1 through 4 is determined by verifying:

- satisfactory completion of an integrity assessment in accordance with Steam Generator Program requirements as part of each SG inspection, and

- plant operation within the operating cycle defined by the

  operational assessment.

Performance Criteria

Accident induced leakage and structural integrity are two of the three performance criteria defined by the Steam Generator Program. These two, along with the third performance criterion, operational LEAKAGE, act together to provide reasonable assurance of tube integrity at normal and accident conditions.

LCO (continued)

The structural integrity and accident induced leakage performance criteria are documented in Specification 5.5.9. The operational LEAKAGE performance criterion is included in LCO 3.4.13, "RCS Operational LEAKAGE." All three performance criteria are described below:

(i) Structural Integrity Criterion
The structural integrity criterion is:

"SG tubing shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown, and all anticipated transients included in the accident analysis design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary to secondary pressure differential and a safety factor of 1.4 against burst applied to the largest primary to secondary pressure differential associated with ASME Section III, Level D service. Additional conditions identified in the design and licensing basis shall be evaluated to determine if the associated loads do not contribute to burst.

Contributing loads that do affect burst shall be assessed with a safety factor of 1.0 and combined with the appropriate load due to the defined pressure differential."

The structural integrity criterion can be broken into two separate considerations:

- Providing a margin of safety against tube burst under normal and accident conditions, and

- Ensuring structural integrity of the SG tubes under all anticipated transients included in the design specification.

Tube Burst

Tube burst is defined as:

"The gross structural failure of the tube wal!. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation."

BASES

LCO (continued)

The structural integrity criterion provides reasonable assurance that a SG tube will not burst during normal or accident conditions. The structural integrity criterion requires that the tubes not burst when subjected to differential pressures equal to 3.0 times those experienced during normal steady state full power operation and 1.4 times ASME Section III, Level D accident pressure differentials. Other loadings required by the design and licensing basis shall be combined with the design basis accident loads without application of the 1.4 safety factor. The safety factors of 3.0 and 1.4 and the requirement to include applicable design basis loads are based on ASME Code Section III Subsection NB (Ref. 5) requirements and Draft Regulatory Guide 1.121 (Ref. 6) guidance.

In the context of the structural integrity criterion, normal steady state full power operation is defined as:

"The conditions existing during MODE 1 operation at the maximum steady state reactor power as defined in the design or equipment specification. Changes in design parameters such as plugging or sleeving levels, primary or secondary modifications, or Thot should be assessed and their effects on differential pressure should be included if significant."

Guidance on accounting for changes in these parameters is provided in the EPRI Steam Generator Integrity Assessment Guidelines (Ref. 7).

In addition to the safety factors of 3.0 and 1.4, further adjustments may be required to ensure representative verification of tube burst integrity for various damage forms. For example, adjustments to include axial loading associated with locked tube supports in recirculating SG designs is addressed in Ref. 8 to ensure that the evaluated or tested conditions are at least as severe as those expected during operating and accident events. However, these loads are not subject to the safety factor applied to normal full power operation and accident pressure differentials.

Tube Structural Integrity

Pursuant to the structural integrity criterion, Ref. 1 requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Secton III, Level A (normal operating conditions) and Level B (upset or abnormal conditions) transients included in the design specification.

(ii) Accident Induced Leakage Criterion
The accident induced leakage criterion is:

"The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 150 gallons per day through each SG for a total of 600 gallons per day through all SGs."

In the context of the accident induced leakage criterion, accident induced leakage rate is defined as:

"Accident induced leakage rate means the primary to secondary LEAKAGE occurring during accidents other than a SG tube rupture when tube structural integrity is assumed. This includes the primary to secondary LEAKAGE rate existing immediately prior to the accident plus additional primary to secondary LEAKAGE induced during the accident."

The accident induced leakage criterion can be broken into two separate considerations:

- Meeting design basis conditions, and

- Limiting accident induced leakage to 150 gallons per day through each SG under all circumstances.

Design Basis

Primary to secondary LEAKAGE is a factor in the activity releases outside containment resulting from a limiting design basis accident. The radiological dose consequences resulting from a potential primary to secondary leak during design basis accidents must not exceed the offsite dose limits required by Ref. 3, or the control room personnel dose limits required by Ref. 2, or the NRC approved licensing basis.

When calculating offsite doses, the safety analysis for the limiting design basis accident, other than a SG tube rupture, sets the initial primary to secondary LEAKAGE in each SG to 150 gallons per day.

LCO (continued)

Limiting Accident Induced Leakage to 150 Gallons per Day through Each SG

Recent experience with degradation mechanisms involving lube cracking has revealed that leakage under accident conditions can exceed the level of operating LEAKAGE by orders of magnitude. Therefore, a separate performance criterion for accident induced leakage was established. The numerical limit for the accident induced leakage criterion is established at the value for operational LEAKAGE (i.e., 150 gallons per day through each SG).

The NRC has concluded (Item Number 3.4 in Attachment 1 to Ref. 8) that additional research is needed to develop an adequate methodology for fully predicting the effects of LEAKAGE on the outcome of some accident sequences. As a result, LEAKAGE greater than the accident induced leakage criterion is not allowed.

(iii) Operational LEAKAGE Criterion

The operational LEAKAGE criterion and its associated Required Action and Surveillance Requirements are contained in LCO 3.4.13, "RCS Operational LEAKAGE." The operational LEAKAGE criterion is not included in the SG Tube Integrity Specification because it is one of the forms of RCS LEAKAGE that are addressed by the RCS Operational LEAKAGE Specification and because, unlike structural integrity and accident induced leakage, it is observable by the operator during MODES 1 through 4. The operational LEAKAGE criterion is presented below for completeness since all of the performance criteria act together to ensure tube integrity.

The operational LEAKAGE criterion is:

"The RCS operational primary to secondary LEAKAGE through any one SG shall be limited to 150 gallons per day"

An explanation of the operational LEAKAGE criterion is provided in the Bases for LCO 3.4.13, 11RCS Operational LEAKAGE."

The Bases for SR 3.4.13.2 indicates that if this SR is not met, compliance with LCO 3.4.18 should be evaluated. If SR 3.4.13.2 is met, then compliance with LCO 3.4.18 need not be evaluated insofar as primary to secondary LEAKAGE is concerned.

APPLICABILITY SG tubes are designed to withstand the stresses due to differential pressures as large as 3.0 times those experienced under normal full power operations or 1.4 times the largest primary to secondary pressure differential for ASME Section III, Level D (faulted) accidents. This requirement is delineated in the structural integrity criterion. This magnitude of differential pressure or the possibility of an accident impacting tube integrity is only possible during MODES 1, 2, 3, and 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1 through 4. When the plant is shut down, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE. In addition, primary coolant activity is also low. Therefore, this LCO is applicable in MODES 1 through 4 only.

ACTIONS The Actions Table is modified by a Note to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each affected tube. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each affected SG tube. The Completion Times of each affected tube evaluation will be tracked separately, starting from the time the Condition was entered.

A.1 and A.2

Condition A applies if it is discovered that one or more inspected SG tubes satisfy the tube repair criteria but were not plugged or repaired in accordance with the Steam Generator Program as required by SR 3.4.18.2. An evaluation of SG tube integrity must be made. SG tube integrity is based on meeting the structural integrity and accident induced leakage performance criteria. In general, an affected tube is one with an indication that satisfies the repair criteria. More information on repair limits is provided in Ref. 8.

If it is discovered that a required plugging or repair was not implemented during a previous inspection, the affected SG tube(s) may have SG tube integrity. In this situation, the SGs were returned to service after the last inspection with a tube already satisfying the repair criteria. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections and still provide assurance that the performance criteria will continue to be met. In order to determine SG tube integrity, an evaluation must be completed that demonstrates that

BASES

ACTIONS (continued)

the performance criteria will continue to be met at the time of the next SG inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered.

A Completion Time of 7 days allows sufficient time to complete the evaluation. If it is determined that tube integrity is not being maintained, Condition B must be entered.

If the evaluation determines that tube integrity is maintained for the affected tube(s), Required Action A.2 allows plant operation to continue until the next outage as long as the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged or repaired prior to entering MODE 4 after the outage. This Completion Time is acceptable since the condition will be corrected no later than at the next inspection of the affected SG and the time to the next inspection is supported by the Steam Generator Program as part of the evaluation completed upon entering Condition A. The timing of the next inspection is based on continuing to meet the structural integrity and accident induced leakage performance criteria.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the factors that tend to challenge tube integrity.

The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower and further deterioration is much less likely.

SURVEILLANCE SR 3.4.18.1
REQUIREMENTS

During shutdown periods the SGs will be inspected as required by the Steam Generator Program. The Steam Generator Program is required by Specification 5.5.9. Ref. 1 and its referenced EPRI Guidelines establish the content of the Steam Generator Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices. During SG inspections the licensee will perform a condition monitoring assessment of the SG tubes. The condition monitoring assessment determines the uas found't condition of the SG tubes following inspection with respect to the structural integrity and accident induced leakage performance criteria. The purpose of Ihe condition monitoring assessment is to ensure that the performance criteria have been met for the previous operating period.

The Steam Generator Program determines the scope of the inspection and the methods used to determine compliance with the performance criteria.

- The inspection scope defines which tubes or areas of tubing within the SG are to be inspected. Inspection scope is a function of existing and potential degradation locations and safety/pressure boundary considerations.

- InspecUon methods are those Non-Destructive Examination (NDE) techniques used to find potential degradation. Inspection methods are a function of degradation morphology, NDE technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.18.1. The Frequency is determined by the operational assessment and other limitations in the PWR Steam Generator Examination Guidelines (Ref. 9). The limitations in Ref. 9 and the operational assessment determine the length of the surveillance period by using information on existing degradations and growth rates to define a cycle length that provides reasonable assurance that the tubing will meet the performance criteria at the next scheduled inspection.

The maximum interval between SG inspections is limited. Catawba will perform required SG inspections of tubing and/or sleeves at intervals no greater than those documented in Specification 5.5.9.

SR 3.4.18.2

During a SG inspection, any inspected tube that satisfies Steam Generator Program repair criteria is repaired or removed from

service by plugging. Repair criteria are defined as:

"Repair criteria are those NDE measured parameters at or beyond which a tube must be repaired using an approved repair method or removed from service by plugging."

The tube repair criteria establish limits for tube degradation that provide reasonable assurance that all tubes left in service (e.g., with degradation not satisfying the repair criteria) will meet the performance criteria at the next scheduled inspection by allowing for anticipated growth during the intervening time interval.

Tube repair criteria are either the standard through wall (Tw) depth based criterion (e.g., 40% Tw for Catawba), or Tw depth based criteria for repair techniques approved by the NRC, or other Alternate Repair Criteria (ARC) approved by the NRC such as a voltage based repair limit per Generic Letter 95-05 (Ref. 10).

The depth based criterion, approved for use at all plants by the NRC, was established when the most frequent form of degradation was general wastage corrosion. This type of degradation structurally bounds other forms of degradation and is characterized by a volumetric loss of the tube wall. This criterion was established to allow for NDE uncertainties and growth and still provide a reasonable assurance that all tubes with degradation not exceeding the criterion will exhibit acceptable structural integrity and accident induced leakage. Additional basis information is provided in Ref. 8.

Since not all forms of tube degradation can be accurately measured for flaw depth in terms of percentage of tube wall thickness, some tubes are "plugged or repaired on detection" to ensure that detected flaws that exceed the depth based criterion are not left in service.

In addition, since the probability of detecting a flaw is not a certainty for a given eddy current technique, it is probable that some flaws will not be detected during an inspection. This condition does not mean that "plug on detection" has not been followed or that the depth based criterion has been violated.

In recent years, improved inspection techniques, knowledge of corrosion mechanisms, and experience have revealed additional types of tube degradation in the form of cracks in the tube wall. In some instances, a reliable method of characterizing specific types of cracks at defined locations within certain SG designs has been

developed. In these cases, the industry has developed, and the NRC has approved ARC to permit leaving a tube in service (as opposed to plugging) when the tube has indications that fall within the limits established by the ARC. "Plug or repair on detection" is not an ARC.

The NRC must approve all repair criteria prior to use. The repair criteria approved for use at Catawba are listed in Specification 5.5.9.

Due to technique and analyst uncertainties, sampling plans, and probability of detection, there is a possibility that tube(s) satisfying the repair criteria will not be detected during a particular SG inspection. If the flaw(s) is detected during a subsequent inspection, the condition is not considered a reportable event unless it is determined that the performance criteria are not met.

SG tube repairs are only performed using approved repair methods. Repair methods are defined as:

"Repair methods are those means used to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. Plugging a SG tube is not a repair."

Repair methods are approved by the NRC either by license amendment or as part of the NRC's approval of applicable ASME Code requirements. The repair methods approved by license amendment (if any) are listed in Specification 5.5.9. The repair methods approved by the NRC through the ASME Code are those specifically listed in ASME Section XI, IWA-4720 (Ref. 11) of Code editions and addenda listed in 10 CFR 50.55a (Ref. 12). New repair methods designed in accordance with general Code requirements (as opposed to being specifically listed in the Code article cited above) may not be implemented without prior NRC approval.

There are no repair methods presently approved by license amendment for use at Catawba.

Inspected SG tubes that satisfy the repair criteria are repaired or removed from service by plugging prior to entry into MODE 4. This is necessary in order to provide reasonable assurance that tube integrity will be maintained until the next scheduled inspection.

REFERENCES 1. NEI 97-06, "Steam Generator Program Guidelines."
2. 10 CFR 50 Appendix A, GDC 19, "Control Room."
3. 10 CFR 100, "Reactor Site Criteria."
4. 10 CFR 50.67, "Accident Source Term."

5. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, "Ru!es for Construction of Nuclear Facility Components, Class 1 Components."

6. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.

7. EPRI Steam Generator Integrity Assessment Guidelines.

8. S.C. Col!ins memo to W.D. Travers, "Steam Generator Action Plan Revision to Address Differing Professional Opinion on Steam Generator Tube Integrity," dated May 11, 2001.

9. EPRI PWR Steam Generator Examination Guidelines.

10. Generic Letter 95-05, "Voltage Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," August 3, 1995.

11. ASME Section Xl, IWA-4720, "Sleeving."

12.10 CFR 50.55a, "Codes and Standards."

5.5.8 Inservice Testing Program

This program provides controls for inservice tesung of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the folloWing:

a. Testing frequencies specified in SecUon Xl of the A~ME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing acUvities	Required Frequendes for performing inservice tesung activities
Weekly	At least once per 7 days
Monthly -	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be constmed to supersede the requirements of any TS.

5.5.9 Steam Generator SG ube urvei ance Pro ram

his program rovides controls for t e inservice inspecti of steam gene tor tubes to ens e that the structural i tegrity of this portio of the RCS is maintained. he program for inse ice inspection of st am generator tu es is based on odification of Regul tory Guide 1.83, R ision 1. The pro ram shall inclu e:

(continued)

Each stea generator shall be determined OPE BLE during shutdown~y selectin nd inspecting at least the minimum of team generators specified in TableS -1.

5.5.9.2 Ste Generator Tube Sam le Selection an Ins ection

e steam generator tube minimum sampl size, inspection result classificatio
n,
nd the corresponding action required sh II be as specified in Table 5.5-2. The inservice inspection of steam generator ubes shall be performed at the frequencies specified in Specification .5.9.3 and the inspected tubes shall be verified acceptable per the acceptan criteria of Specification 5.5.9.4. The tub
es
selected for each inservice inspecti shall include at least 3% of the total number of tubes in all steam gene tors; the tubes selected for these inspecti
on
shall be selected on a random b is except:

a. Where experience in 5 ilar plants with similar water chemistry mdi tes critical areas to be in ected, then at least 50% of the tubes inspe ed shall be from these tical areas;
b. The first sample   tubes selected for each inservice inspectio (subsequent to   e preservice inspection) of each steam gen rator shall include:

1. All n nplugged tubes that previously had detecta e wall pe trations (greater than 20%),

2. T bes in those areas where e~~erience has dicated potential roblems, and

3. A tube inspection (pursuant to Specifica on 5.5.9.4.a.8) shall be performed on each selected tube. If a selected tube does not permit the passage of the eddy curre probe for a tube inspection, this shall be recorded a an adjacent tube shall be selected and subjected to a tube i pection.

c. The tubes selected as the second and ird samples (if required by Table 5.5-2) during each inservice in ection may be subjected to a partial tube inspection provided:

1. The tubes selected for th se samples include the tubes from those areas of the tube heet array where tubes with imperfections were pr iously found, and

(continued)

0

The inspections indude those portions 0 he tubes where imperfections were previously found.

Th results of each sample inspection shall be da (fled into one of the followi
ng

t ee categories:

Category

Less than 50 of the total tubes inspected are degraded es and none of the inspected tubes aredefe ive.

c2One more tubes1 but not niore than 1% of the tot tubes inspected are defective, or between 5% a 10% of the total tubes inspected are degraded

0-3 More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspe ~ons, previously degraded tubes must exhibit significan (greater t n 10%) further wall penetrations to be included in the ove percent ge calculations.

5.5.9.3 Ins ectio re uencies

The a ve required inservice inspections of steam generator tu 5 shall be ped med at the following frequencies:

a The first inservice inspection after steam generator r placement shall be performed after at least 6 Effective Full Power Mo 5 but within 24 calendar months of initial criticality after steam g erator replacement (Unit 1). The first inservice inspection shall be rformed after 6 Effective Full Power Months but within 24 calendar mo 5 of initial criticality (Unit 2). Subsequent inservice inspections shall performed at intervals of not less than 12 nor more than 24 calend months after the previous inspection. If two consecutive inspectio , not including the preservice inspection, result in all inspection result falling into the C-i category or if

(continued)

b. If th results of the inservice inspection of a steam enerator conducted in ccordance with Table 5.5-2 at 40-month inte Is fall in Category ~3, e inspection frequency shall be increased to least once per 20 months. The increase in inspection frequen shall apply until the subsequent inspections satisfy the criteria o pecification 5.5.9.3.a; the interval may then be extended to a maxim of once per 40 months; and

c. Additional1 unscheduled inservice insp tions shall be performed on each steam generator in accordance with e first sample inspection specified in Table 5.5-2 during the shutdo subsequent to any of the following conditions:

1. Reactor-to-seconda tubes leaks (not including leaks originating from tube-to-tube eet welds) in excess of the limits of Specification 3.4 3,
2. A seismic rrence greater than the Operating Basis Earthquak
3. A loss ~coolant accident requiring actuation of the Engineered Safe Features1 or
4. A am steam line or feedwater line break.

The provisio of SR 3.0.2 are applicable to the SG Tube Surveillance rogram

5.5.9.4 Acce ance Criteria

a. As used in this specification:

1. Imnerfection means an exception to the Imensions1 finish or contour of a tube from that req uired b abrication drawings or specifications. Eddy~urrent testing dications below 20% of the nominal tube wall thickness, if det table1 may be considered as imperfections;

2. Degradation means a servic -induced cracking1 wastage, wear or general corrosion occurri on either inside or outside of a tube;

(continued)

5.5.9.4 Acce tanc Griteria (continued)

Degraded Tube means a tube containin imperfections greater than or eq ual to 20% of the nominal t e wall thickness caused by degradation;

4. % degradation means the perc tage of the tube wall thickness affected or removed by degr ation;

5. Defect means an imperf tion of such severity that it exceeds the plugging limit. A tube ntaining a defect is defective;

6. Plugging Limit m ns the imperfection depth at or beyond which the tube shall b removed from service by plugging. The plugging limit is equal 40% of the nominal tube wall thickness.

7. Unservic ble describes the condition of a tube if it leaks or contai a defect large enough to affect its structural integrily in

the e nt of an Operating Basis Earthquake, a lO5s-of~oolant ac dent, or a steam line or feedwater line break as specifie in .9.3.c, above;

8. Tube InsDection means an inspection of the steam ge rator tube from the point of entry completely around the U-ben to the point of exit; and

9. Preservice InsDection means an inspection of e full length of each tube in each steam generator perform d by eddy current techniques prior to service to establish a seline condition of the tubing. This inspection shall be perfo d prior to initial POWER OPERATION using the equipment an techniques expected to be used during subsequent inservice i pections.

b. The steam generator shall be determ ed OPERABLE after completing the corresponding actions required y Table 5.5-2.

(continued)

Preservice Ins tion	No	Yes
No. of m Generators per Unit		
First Inservice Inspection (Unit 2)		
Second & Subsequent Inservice Inspections	One1	One2

Table Notation

1.The inservice inspection ay be limited to one steam generator on a rotating Sc
edule
encompassing 3 N % the tubes (where N is the number of steam generator in th

e unit) if the
results of the first or evious inspections indicate that all steam generators e perf orming in a (ike
manner. Note that der some circumstances, the operating conditions in ne or m ore steam
generators may b found to be more severe than those in other steam g erators. Under such
circumstances t sample sequence shall be modified to inspect the m t severe c onditions.

2. Each of the ot er two steam generators not inspected during the firs nservice in spection after the
steam gene tor replacement shall be inspected during the second nd third inspe ctions (Unit 1).
Each of th other two steam generators not inspected during the f t inservice ins pection shall be
inspect during the second and third inspections (Unit 2). The urth and subsequ ent inspections

JN~~TA
Gatawba Units 1 and 2 5.5-11 Amendment Nos.~5
1ST SAMPLE INSPECTION 2ND SAMPLE INSPECTION 3RD SAMPL E INSP ION
Sample Size Result ion Result Action Required Result Act Required
equired
A minimum CI None N/A N/A
of S tubes
perSG
C-2 Plug CI None
defective
tubes and
inspect
addWonal 2S
tubes in tnis
SG
C2 Plug efective CI None
C2 Plug defective tubes
C3Pedonn action for C3

result of first sample
CS Perform action for N/A N/A
CS result of first
sample
CS Ins Ct all All other None WA WA

tubes in each
other SG.
Prompt
notification to
NRC
pursuant to
1 OCFR50.72
(b)(2)
Some SGs Perform action for N/A WA
C2 but no C2 result of
additional second sample
SGs are
CS

Additional Inspect all tubes in N/A N/A
SG is CS each SG and p1
defective tube
Notification NRC
pursuant
1 OCFR 72 (b)(2)

S SN/n % Where N is the number of steam gene tors in the unit, and n is the nu
mber of steam f
generators inspected during an insp tion
Gatawba Units 1 and 2 5~5-12 Amendment Nos~~5
INSERT A for TS 5.5.9, Steam Generator (SG) Program:

A Steam Generator Program shall be established and implemented to ensure that SG
tube integrity is maintained, and to describe SG condition monitoring, performance
criteria, repair methods, repair criteria, and inspection intervals. The Steam Generator

Program shall address the following topics:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with re~pect to the performance criteria for ~tructural and accident induced leakage integrity. The "as found" condition refers to the condition of the tubing during a SG inspection outage, as determined from the inservice inspection results or b

y

other means, prior to the plugging or repair of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tube

s

are inspected, plugged, or repaired to confirm that the performance criteria are being met.

b. Provisions for verifying SG tube integrity. SG tube integrity is maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: SG tubing shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown, and all anticipated transients included in the accident analysis design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary to secondary pressure differential and a safety factor of 1.4 against burst applied to the largest primary to secondary pressure differential associated with ASME Section III, Level D service. Additional conditions identified in the design and licensing basis shall be evaluated to determine if the associated loads do not contribute to burst. Contributing loads that do affect burst shall be assessed with a safety factor of 1.0 and combined with the appropriate load due to the defined pressure differential.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 150 gallons per day through each SG for a total of 600 gallons per day through all SGs.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged or repaired prior to entry into MODE 4.

d. Provisions for SG tube repair methods. SG tube repair methods shall pro vide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of this program, tube pluggin g is not a repair. Acceptable tube repair methods are those designs specifically listed in ASME Section XI, IWA-4720. NRC endorsement of the applicable Code sections is required prior to use. Endorsement is documented in 10 CFR 50.55a .

e. Provisions for SG tube inspection intervals. SG tube inspection intervals shall be established based on the following:

1. No SG with Alloy 600 thermally treated tubing shall operate more than 48 effective full power months without being inspected.

2. No SG with Alloy 690 thermally treated tubing shall operate more than 72 effective full power months without being inspected.

3. If indications of corrosion cracking are found during a SG inspection, then perform an inspection of that SG at the next refueling outage or within 24 effective full power months, whichever is less.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

14. DPC-NE-2009-P-A, "Westinghouse Fuel Transition Report" (DPC Proprietary).

15. WCAP-1 2945-P-A, Volume I and Volumes 2-5, "Code Qualification Document for Best-Estimate Loss of Coolant Analysis" LW Proprietary).

The COLR will contain the complete identification for each of the Technical Specifications referenced topical reports used to prepare the COLR (i.e.1 report number, title1 revision number, report date or NRC SER date, and any supplements).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be

provided upon issuance for each reload cycle to the NRC.

5.6.6 Ventilation Systems Heater ReDort

When a report is required by LCO 3.6.10, "Annulus Ventilation System (AVS),
"
LCO 3.7.10, "Control Room Area Ventilation System (CRAVS)," LCO 3.7.12, Auxiliary Building Filtered VenUlation Exhaust System (ABFVES)," LCO 3.7.1
3,
"Fuel Handling Ventilation Exhaust System (FHVES)," or LCO 3.9.3, "Containment Penetrations," a report shall be submitted within the following
30
days. The report shall outline the reason for the inoperability and the planne
d
actions to return the systems to OPERABLE status.

5.6.7 PAM Re~ort

When a report is required by LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. T
he
report shall outline the preplanned alternate method of monitoring, the caus
e of
the inoperability, and the plans and schedule for restoring the instrumentatio
n
channels of the Function to OPERABLE status.

5.6.8 Steam Generat7~~ube Ins~ection Re~ort

a. The number of tube plugged in each stea generator shall be ported to the NRC within days following compl on of the program

(continued)

5.6 Reporting Requirements

5.6.8 Steam Generator Tube Ins~ection Re~rt (Continued)

b. The co plete results of the Steam Generat r Tube Surveillance Program shall b reported to the NRC within 12 mo ths following the completion o the p ram and shall include:

1 - Number and extent of tubes in ected,

Location and percent of wall- ickness penetration for each

indication of an imperfectio and

3. Identification of tubes plu ged.

The results of inspections of S am generator tubes which fall to Category 03 shall be report to the NRC within 30 days pri r to the restart of the unit following e inspection. This report shall rovide a description of the tube de adation and corrective measur taken to prevent recurrence.

INSERT B for TS 5.6.8, Steam Generator (SG) Tube Inspection Report:

If the results of the SG inspection indicate greater than 1% of the inspected tubes i n any SG exceed the SG tube repair criteria specified in Specification 5.5.9, "Steam Gene rator (SG) Program," a report shall be submitted within 120 days after the initial entry int o MODE 4 following completion of the inspection. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanis
m,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged or repaired during the inspection outage for each act
ive

degradation mechanism,

f. Repair method utilized and the number of tubes repaired by each repair method
,

g. Total number and percentage of tubes plugged and/or repaired to date,

h. The effective plugging percentage for all plugging and tube repairs in each SG, and

I. The results of condition monitoring, including the results of tube pulls and in-si
tu

testing.

ATTAC~Th:NT 2

REPRINTED TS AND BASES PAGES FOR CATAWBA
(TO BE PROVIDED TO NRC UPON ISSUANCE OF ANENDMENTS)

ATTACHMENT 3

BACKGROUND, DESCRIPTION OF THE PROPOSED CHANGES, AND TECHNICAL JUSTIFICATION

A. Introduction

In December of 1998, the NRC Staff acknowledged that the Steam Generator Program described by NEI 97-06 and its referenced EPRI Guidelines provides an acceptable starting point to use in the resolution of differences between it and the staff's proposed Generic Letter and draft Regulatory Guide (DG-I074). Since then the industry and the NRC have participated in a series of meetings to resolve the differences and develop the regulatory framework necessary to implement a comprehensive Steam Generator Program. This license amendment request is the culmination of that effort.

B. Background

The steam generator (SG) tubes in pressurized water reactors have a number of important safety functions. SG tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the RCPB, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes also isolate the radioactive fission products in the primary coolant from the secondary system. SG tube integrity is necessary in order to satisfy the tubing's safety functions. Maintaining tube integrity ensures that the tubes are capable of performing their intended safety functions consistent with their licensing basis, including applicable regulatory requirements.

Concerns relating to the integrity of the tubing stem from the fact that the SG tubing is subject to a variety of degradation mechanisms. SG tubes have experienced tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. When the degradation of the tube wall reaches a prescribed repair criterion, the tube is considered defective and corrective action is taken.

The criteria governing structural integrity of SG tubes were developed in the 1970s from assumptions relative to

uniform tube wall thinning. This led to the establishment of a through wall SG tube repair criterion (e.g., 40%) that has historically been incorporated into most pressurized water reactor TS and has been applied, in the absence of other repair criteria, to all forms of SG tube degradation where sizing techniques are available. Since the basis of the through wall depth criterion was 3600 wastage, it is generally considered to be conservative for other mechanisms of SG tube degradation. The repair criterion does not allow licensees the flexibility to manage different types of SG tube degradation. Licensees must either use the through wall criterion for all forms of degradation or obtain approval for use of more appropriate repair criteria that consider the structural integrity implications of the given mechanism.

For the last several years, the industry, through the EPRI Steam Generator Management Program (SGMP), has developed a generic approach to improving SG performance referred to as "Steam Generator Degradation Specific Management" (SGDSM). Under this approach, different methods of inspection and different repair criteria may be developed for different types of degradation. A degradation specific approach to managing SG tube integrity has several important benefits. These include:

- improved scope and methods for SG inspection,
- industry incentive to continue to improve inspection methods, and
- development of plugging and repair criteria based on appropriate Non-Destructive Examination (NDE) parameters.

As a result, the assurance of SG tube integrity is improved and unnecessary conservatism is eliminated.

Over the course of this effort, the SGMP has developed a series of EPRI guidelines that define the elements of a successful SG program. These guidelines cover topics such as:

- SG examination [1]

• SG integrity assessment [2]

• in-situ pressure testing [3]

• primary to secondary leakage [4]

• primary water chemistry [5], and

• secondary water chemistry [6]

These EPRI guidelines, along with the upper tier document (NEI 97-06, "Steam Generator Program Guidelines" [7]) that ties the entire SG program together, define a comprehensive, performance based approach to managing SG performance.

Revising the existing regulatory framework to accommodate degradation specific management is the most appropriate way to address the issues of regulatory stability, resource expenditure, use of state-of-the-art inservice inspection techniques, repair criteria, and enforceability. The NRC staff has stated that an integrated approach for addressing SG tube integrity is essential and that materials, systems, and radiological issues that pertain to tube integrity need to be considered in the development of the new regulatory framework.

This license amendment request provides the integrated approach for addressing SG tube integrity.

C. Description of Amendment Request

The proposed amendments replace the SG detailed programmatic requirements contained in TS 5.5.9 with a SG Thbe Integrity TS (TS 3.4.18) and Bases and revises the TS for RCS Operational Leakage (TS 3.4.13), SG Tube Surveillance Program (TS 5.5.9), and SG Tube Inspection Report (TS 5.6.8).

Marked-up and new TS and Bases pages are in Attachment 1.

The TS for SG Thbe Integrity contains surveillance requirements for tube integrity verification and repair and actions necessary should tube integrity not be maintained. The SG Tube Integrity TS Bases describes the SG performance criteria and inspection intervals defined in the SG Program. Changes to the SG Tube Integrity TS Bases will be governed by the TS Bases Control Program under the provisions of 10 CFR 50.59.

The changes to TS 3.4.13 reference the plant's SG Program required by TS 5.5.9 for the surveillance requirements necessary to verify primary to secondary leakage. In addition, the actions are changed to treat primary to secondary leakage the same as RCS pressure boundary

leakage; shutdown must be commenced upon exceeding a leakage rate of 150 gallons per day in any SG. The previous TS allowed 4 hours to reduce leakage to within limits. Finally, one of the surveillance requirements is revised to include a note that clarifies applicability requirements. The changes to TS 5.5.9 require the implementation of a SG Program and describe key aspects of the program. This section also documents the SG performance criteria, plugging criteria, repair methods, and maximum inspection intervals. The change to TS 5.6.8 defines the requirement for, and contents of, the SG tube inspection report.

The combination of these changes will implement the regulatory aspects of NEI 97-06, '1Steam Generator Program Guidelines".

D. Description of Proposed Changes to SG Requirements

The most obvious changes in the proposed TS are the replacement of the detailed prescriptive requirements in TS 5.5.9 by proposed TS 3.4.18 and the modification of TS 5.5.9. The new SG TS replaces a large amount of prescriptive, outdated details on SG inspection requirements with a requirement to implement a state of the art performance based program that is supported by a NEI SG initiative, extensive industry guidance, and an active industry technical advisory group. The revised administrative TS add more detail on the required elements of a Steam Generator Program and also refer to the industry program requirements. These changes are a significant improvement over the existing outdated TS requirements.

The details of the proposed changes are delineated in the following table that summarizes SG operation under the current licensing basis and under the Steam Generator Program required by the proposed license amendments. Note that many of the requirements discussed in the following section are part of the Steam Generator Program and are not specifically included in the TS. The location of the requirement is provided in the third column of the table.

Throughout the table "Reg" means regulations, "TS'1 means Technical Specifications, "TSB" means TS Bases, and "SGP" means Steam Generator Program.

Attachment 3 Page 4

Condition or Current Location - Note
Requirement Licensing Basis Proposed Change

Operational 576 gallons per Operational leakage
primary to day total irS - < 150 gallons
secondary through all SGs per day through any
leakage and 150 gallons one SG
per day through
any one SG

RCS leakage Reduce leakage Operational leakage 2
not withinto within irS - Be in Mode 3 in
limits limits in 4 6 hours and in Mode

hours or be in 5 in 36 hours
Mode 3 in 6
hours and in
Mode 5 in 36
hours
RCS leakage Note states: Operational leakage 3
determined Not required to irS new Notes:
by water be performed in
inventory Mode 3 or 4 1. Made editorial
balance until 12 hours changes to Note
of steady state
operation 2. Made SR not
applicable to
primary to secondary
leakage
SG tube Verify in Operational leakage 4
integrity accordance with irS - Restrict the SR
verification the SG Tube to primary to
Surveillance secondary leakage
Program determination
Frequency of 6 to 40 months SG tube integrity irs 5
verification depending upon - Requires SR
of tube SG category frequency in
integrity defined by accordance with the _________

Attachment 3 Page 5
previous SGP
inspection
results SGP - Frequency is
dependent upon
tubing material and
the previous
inspection results
and the anticipated
defect growth rate

Administrative TS -
Establishes maximum
inspection intervals
Tube sample Based upon SG SGP - Dependent upon 6
selection category, a pre-outage
industry evaluation of actual

experience, degradation
random locations and
selection, mechanisms, and
existing operating experience
indications, - 20% of all tubes
and resultsof as a minimum
the initial
sample set - 3%
of all tubes as
a minimum

Inspection Not specified SG tube integrity TS 7
techniques - SR 3.4.18.1
requires that tube
integrity be
maintained in
accordance with the
SGP

SGP - Establishes
requirements for
qualifying NDE
techniques/Requires
use of qualified
techniques in SG
inspections/Requires
a pre-outage
evaluation of

Attachment 3 Page 6
potential tube
degradation
morphologies and
locations and an
identification of
NDE techniques
capable of finding
the degradation
Inspection Hot leg point SGP - Inspection 8
scope of entry to scope is defined by
(typically) the the degradation

first support assessment that
plate on the considers existing
cold leg side and potential
of the U-bend degradation
morphologies and
locations
Performance Operational Operational leakage 9
criteria leakage - < 576 TS - Operational
gallons per day leakage < 150
through all SGs gallons per day
and < 150 through.any one SG
gallons per day
through any one SG tube integrity TS
SG - Requires that tube
integrity be
No criteria maintained
specified for
structural Administrative ~s -
integrity or Documents structural
accident integrity and
induced leakage accident induced
leakage performance
criteria which are
dependent upon
design basis
limits/Requires
condition monitoring
assessment to verify
compliance

TSB - Relates tube
_________________________________ integrity to

Attachment 3 Page 7
satisfaction of the
performance criteria

Repair Plug tubes with Administrative TS - 10
criteria imperfections Criteria unchanged
extending > 40%

through wall

Approved
criteria listed
in the TS
Failure to Performance Operational leakage 11
meet criteria not and SG tube
performance defined/Primary integrity TS -
or repair to secondary Contains primary to
criteria leakage limit secondary leakage
and actions limit, SG tube
included in the integrity
TS requirements and
actions required
Plug tubes upon failure to meet
exceeding performance criteria
repair criteria
Plug or repair tubes
exceeding repair
criteria

Repair Methods (except Administrative TS - 12
methods plugging) Methods (except
require plugging) require
previous NRC
approval by the approval/Approved
NRC/Approved methods listed
methods listed either in the
in TS (no administrative TS
approved (no methods proposed
methods for Catawba) or in
applicable for the ASME Code
Catawba at
present)
ReportingPlugging report Reg - NRC reports 13
Attachment 3 Page 8
requirements required 15 required in
days after each accordance with 10
inservice CFR 50.72 and 10 CFR
inspection, 12- 50.73 upon failure
month report to meet a
documenting performance

	inspection results, and reports in accordance with 10 CFR 50.72 when the inspection results fall into category C-3	criterion Administrative TS - 120 days after the initial entry into Mode 4 if > 1% of the inspected tubes in any affected SG exceed repair criteria	
Definitions	Normal TS definitions did not address SGP issues	SGP/TSB - Includes applicable SGP definitions	14

Further explanation of the information presented in the table above is provided in the following notes referenced in the table.

Note 1. Operational Leakage

The primary to secondary leakage limit provides assurance against tube rupture at normal operating and faulted conditions. This together with the allowable accident induced leakage limit helps to ensure that the dose contribution from tube leakage will be limited to less than the 10 CFR 100 and GDC 19 dose limits or other NRC approved licensing basis (e.g., 10 CFR 50.67) for postulated faulted events.

This limit also contributes to meeting the GDC 14 requirement that the reactor coolant pressure boundary "have an extremely low probability of abnormal leakage, of rapidly propagating to failure, and of gross rupture." The proposed TS references the SG Program for surveillance requirements. The SG Program uses the EPRI primary to secondary leak guidelines [4] to establish sampling requirements for determining primary to secondary leakage and plant shutdown requirements if leakage limits are exceeded. The guidelines ensure leakage is effectively monitored and timely action is taken before a leaking tube exceeds the performance criteria.

The TS requirement to limit primary to secondary leakage through any one SC to 150 gallons per day is based on an initial condition of the safety analysis. The dose analyses of record for Catawba assume a primary to secondary leakage through each SC of 150 gallons per day, for a total of 600 gallons per day through all SGs. Catawba's dose analyses no longer assume a total primary to secondary leakage through all SGs of 576 gallons per day; therefore, the 576 gallons per day limit is being deleted by these amendment requests. The value of 576 gallons per day was introduced into the Catawba TS as a result of License Amendments 102 and 96 for Units 1 and 2, respectively. Amendments 102 and 96 implemented the interim SC tube plugging criteria prior to the replacement of the Unit 1 SGs. Since the Unit 1 SGs have subsequently been replaced, the 576 gallons per day limit is no longer necessary. The assumption of 150 gallons per day through each SC bounds the 576 gallons per day limit currently contained in TS 3.4.13.d.

Note 2. Operational Leakage Actions
If primary to secondary leakage exceeds 150 gallons per day, plant shutdown must be commenced. Node 3 must be achieved in 6 hours and Node 5 in 36 hours. The existing TS allow 4 hours to reduce primary to secondary leakage to less than the limit. The proposed TS removes this allowance.

The removal of the 4-hour period during which primary to secondary leakage can be reduced to avoid a plant shutdown results in a TS that is significantly more conservative than the existing operational leakage TS. This change is consistent with the SC Program that also does not allow 4 hours before commencing a plant shutdown.

Note 3. RCS Leakage Determined by Water Inventory Balance
The existing operational leakage SR 3.4.13.1 (RCS water inventory balance) contains a note that states that the surveillance is not required in Mode 3 or 4 until 12 hours

of steady state operation. The proposed amendments change this note to state: '1Not required to be performed until 12 hours after establishment of steady state operation." This

change was already approved for Revision 2 of the Westinghouse Improved TS, but was inadvertently overlooked.

The proposed TS adds a second note that makes the water inventory balance method not applicable to primary to secondary leakage determination. This change is proposed because leakage of 150 gallons per day or lower cannot be measured accurately by an RCS water inventory balance. This change is necessary to make the SR appropriate for the proposed LCO.

Note 4. SG Thbe Integrity Verification
SR 3.4.13.2 in the existing TS requires verification of tube integrity in accordance with the SG Tube Surveillance Program. This surveillance is no longer appropriate since tube integrity is addressed by newly proposed TS 3.4.18. TS 3.4.13 now applies only to primary to secondary leakage. SR 3.4.13.2 has been changed to verify the LCO requirement on primary to secondary leakage only. SG tube integrity is verified through SR 3.4.18.1.

The SG Program and the EPRI PWR primary to secondary leak guidelines [4] provide guidance on leak rate monitoring.

The proposed surveillance frequency is determined by the SG Program requirements. The SG Program's primary to secondary leakage test frequencies are based upon guidance provided in the EPRI PWR primary to secondary leak guidelines. During normal operation the program depends upon continuous process radiation monitors and/or radiochemical grab sampling. The monitoring and sampling frequency increases as the amount of detected leakage increases or if there are no continuous process radiation monitors available.

Note 5. Frequency of Verification of Tube Integrity
The existing TS contain prescriptive inspection intervals which depend on the condition of the tubes as determined by the last SG inspection. The tube condition is classified into one of three categories based on the number of tubes found degraded and defective. The minimum inspection interval is no less than 12 and no more than 24 months unless the results of two consecutive inspections are in the best category (no additional degradation), and then the interval can be extended to 40 months.

The surveillance frequency in proposed TS 3.4.18 is governed by the requirements in the SG Program and specifically by References [1) and (2]. The proposed frequency is also prescriptive, but has a stronger engineering basis than the existing TS requirements. The interval is dependent on tubing material and whether any active degradation is found. The interval is limited by existing and potential degradation mechanisms and their anticipated growth rate. In addition, a maximum inspection interval is established by administrative TS 5.5.9.

The maximum inspection interval for Alloy 600 thermally treated tubing is 48 EFPM. Even though the maximum interval is slightly longer than allowed by current TS, it is only applicable to SGs with advanced materials; it is only achievable early in SG life and only if the SGs are free from active degradation. In addition, the interval must be supported by an evaluation that shows that the performance criteria will continue to be met at the next SG inspection. Taken in total, the proposed inspection intervals provide a larger margin of safety than the existing requirements because they are based on an engineering evaluation of the tubing condition and potential degradation mechanisms and growth rates, not only on the previous inspection results. As an added safety measure, the minimum sample size inspected at each interval is significantly larger than that required by current TS

(20% versus 3%); thus prov~ding added assurance that any degradation within the SGs will be detected and accounted for in establishing the inspection interval.

The maximum inspection interval for Alloy 690 thermally treated tubing is 72 EFPM. Even though the maximum interval is longer than allowed by current TS, it is only applicable to SGs with advanced materials; it is only achievable early in SG life and only if the SGs are free from active degradation. In addition the interval must be supported by an evaluation that shows that the performance criteria will continue to be met at the next SG inspection. Taken in total, the proposed inspection intervals provide a larger margin of safety than the existing requirements because they are based on an engineering evaluation of the tubing condition and potential degradation mechanisms and growth rates, not only on the previous inspection results. As an added safety measure, the SGP requires a minimum sample size at each inspection that is significantly larger than that required by current TS (20% versus 3%); thus

providing added assurance that any degradation within the SGs will be detected and accounted for in establishing the inspection interval.

The proposed maximum inspection intervals are based on the historical performance of advanced SG tubing materials. Reference 11 shows that the performance of Alloy 600 thermally treated tubing and Alloy 690 thermally treated tubing is significantly better than the performance of Alloy 600 mill annealed tubing, the material used in many plants' SG tubing at the time that many plants' TS were written. There has been only one instance of cracking in Alloy 600 thermally treated tubes in a U.S. SG and this degradation appears to be limited to a small number of tubes in specific SGs that were left with high residual stress as a result of a problem in their manufacturing process. The mechanism is not a result of operational degradation. There are no known instances of cracking in Alloy 690 thermally treated tubes in either U.S. or international SGs.

In summary, the proposed amendments are an improvement over the existing TS. The existing TS bases inspection intervals on the results of previous inspections; it does not require an evaluation of expected performance. The proposed TS uses information from previous inspections at Catawba as well as industry experience to evaluate the length of time that the SGs can be operated and still provide reasonable assurance that the performance criteria will be met at the next inspection. The actual interval is the shorter of the evaluation results and the requirements in Reference 2. Allowing plants to use the proposed inspection intervals maximizes the potential that plants will use improved techniques and knowledge since better knowledge of SG conditions supports longer intervals.

Note 6. Tube Sample Selection
The existing TS base tube selection on SG conditions and industry and plant experience. The minimum sample size is 3% of the tubes. The proposed TS refers to the SGP degradation assessment guidance for sampling requirements. 'The minimum sample size is 20% of the tubes.

The SGP requires the preparation of a degradation assessment (DA) before every SG inspection. The DA is the key document used for planning a SG inspection, where inspection plans and related actions are determined,

documented, and communicated prior to the outage. The DA addresses the various reactor coolant pressure boundary components within the SG (e.g., plugs, sleeves, tubes, and components that support the pressure boundary.) In a DA, tube sample selection is performance based and is dependent upon actual SG conditions and operational experience at Catawba and of the industry in general. Existing and potential degradation mechanisms and their locations are evaluated to determine which tubes will be inspected. Tube sample selection is adjusted to minimize the possibility that tube integrity might degrade during an operating cycle beyond the limits defined by the performance criteria. The EPRI SG examination guidelines [1] and EPRI SG integrity assessment guidelines [2] provide guidance on degradation assessment.

In general, the sample selection considerations required by the existing TS and the requirements in the SGP as proposed by these license amendments are consistent, but the SGP provides more guidance on selection methodologies and incorporation of industry experience and requires more extensive documentation of the results. Therefore, the sample selection method proposed in these amendment requests is more conservative than the existing TS requirements. In addition, the minimum sample size in the proposed requirements is larger.

Note 7. Inspection Techniques
Proposed SR 3.4.18.1 requires that tube integrity be verified in accordance with the requirements of the SGP. The SGP uses Reference [1] to establish requirements for qualifying NDE techniques and maintains a list of qualified techniques and their capabilities.

The SGP requires the performance of a DA before every SG inspection and refers utilities to EPRI SG examination guidelines [1] and EPRI SG integrity assessment guidelines [2) for guidance on its performance. The DA will identify current and potential new degradation locations and mechanisms and NDE techniques that are effective in detecting their existence. Tube inspection techniques are chosen to reliably detect flaws that might progress during an operating cycle beyond the limits defined by the performance criteria.

The current TS contain no requirements on NDE inspection techniques.

This change is an improvement over the existing TS that contained no similar requirement.

Note 8. Inspection Scope
The existing TS include a definition of inspection that specifies the end points of the eddy current examination of each tube. Typically an inspection is required from the point of entry of the tube on the hot leg side to some point on the cold leg side of the tube, usually at the first tube support plate after the U-bend. This definition is overly prescriptive and simplistic and has led to interpretation questions in the past.

The proposed TS do not include a definition for inspection nor does it provide prescriptive guidance for determining inspection scope; instead, SR 3.4.18.1 refers to the SGP for the conduct of inspections. The SGP provides extensive guidance and a defined process, the DA, for determining the extent of a tube inspection. This guidance takes into account industry and plant specific history to determine potential degradation mechanisms and the location that they might occur within the SG. This information is used to define a performance based inspection scope targeted on plant specific conditions and SG design.

The proposed change is an improvement over the existing TS because it focuses the inspection effort on the areas of concern, thereby minimizing the unnecessary data that the NDE analyst must review to identify indication of tube degradation.

Note 9. Performance Criteria
The proposed amendments provide performance based regulatory oversight of the SGP. A performance based approach has the following attributes:

- measurable parameters,

- objective criteria to assess performance based on risk insights,

- deterministic analysis and/or performance history, and

- licensee flexibility to determine how to meet established performance criteria.

The performance criteria used for SGs are based on tube structural integrity, accident induced leakage, and operational leakage. The structural and accident induced leakage criteria were developed deterministically and are

consistent with the Catawba licensing basis. The operational leakage criterion was based on providing added assurance against tube rupture at normal operating and faulted conditions. The proposed structural integrity and accident induced leakage performance criteria are new requirements. The structural'integrity and accident induced leakage performance criteria are documented in Administrative TS 5.5.9. The requirements and methodologies established to meet the performance criteria are documented in the SGP. The existing TS contain only the operational leakage criterion; therefore, the proposed change is more conservative than the existing requirements.

The SG performance criteria identify the standards against which performance is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining RCPB integrity throughout each operating cycle.

The structural integrity performance criterion is:

SG tubing shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown, and all anticipated transients included in the accident analysis design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary to secondary pressure differential and a safety factor of 1.4 against burst applied to the largest primary to secondary pressure differential associated with ASME Section III, Level D service. Additional conditions identified in the design and licensing basis shall be evaluated to determine if the associated loads do not contribute to burst. Contributing loads that do affect burst shall be assessed with a safety factor of 1.0 and combined with the appropriate load due to the defined pressure differential.

The structural integrity performance criterion is based on providing reasonable assurance that a SG tube will not burst during normal operation or postulated accident conditions. In addition, the structural integrity criterion requires that the primary membrane stress intensity in a tube not exceed the yield strength for Service Level A (normal conditions) and Service Level B

(upset conditions) transients included in the design specification.

The structural integrity performance criterion includes safety factors of 3.0 and 1.4 against burst for specific differential pressure conditions. These safety factors are design basis parameters specified by the ASME Code and used to determine the minimum thickness of an undegraded SG tube.

In addition to the safety factors of 3.0 and 1.4, the performance criterion requires further adjustments to ensure representative verification of tube burst integrity for various damage forms. The assessment of these additional conditions as defined in the design and licensing basis, assures that other loading conditions that can significantly contribute to tube burst are addressed. Such loads include loads associated with locked tube supports which could be postulated to develop in recirculating SG designs. The inclusion of these loads, when determined to affect tube burst conditions, shall have a safety factor of 1.0 applied to the appropriate load value.

Adjustments to include contributing loads are addressed in the applicable EPRI guidelines to ensure that the evaluated or tested conditions are at least as severe as those expected during normal operating conditions and Level D accident events.

An explanation of the structural integrity performance criterion is provided in the Bases for the Steam Generator Tube Integrity TS.

The accident induced leakage performance criterion is:

The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 150 gallons per day through each SG for a total of 600 gallons per day through all SGs.

Primary to secondary leakage is a factor in the activity releases outside containment resulting from a limiting design basis accident. The potential dose consequences

from primary to secondary leakage during postulated design basis accidents must not exceed the radiological limits imposed by 10 CFR Fart 100 guidelines1 or the radiological limits to control room personnel imposed by GDC 19, or other NRC approved licensing basis.

When calculating of fsite doses, the safety analysis for the limiting design basis accident, other than a SG tube rupture, sets the initial primary to secondary leakage in each SG to 150 gallons per day. Recent experience with degradation mechanisms involving tube cracking has revealed that leakage under accident conditions can exceed the level of operating leakage by orders of magnitude. Therefore, a separate performance criterion for accident induced leakage was established. The numerical limit for the accident induced leakage criterion is established at the value for operational leakage (i.e., 150 gallons per day through each SG).

An explanation of the accident induced leakage performance

criterion is provided in the Bases for the Steam Generator
Thbe Integrity TS.

The operational leakage performance criterion is:

The RCS operational prixnary to secondary leakage through any one SG shall be ijini ted to 150 gallons per day.

Plant shutdown should commence if primary to secondary leakage exceeds 150 gallons per day from any one SG.

The operational leakage performance criterion is documented in the RCS Operational Leakage TS 3.4.13.

The proposed Administrative TS that contains the structural integrity and accident induced leakage performance criteria (5.5.9) is more conservative than the existing TS. The existing TS do not address the structural integrity and accident induced leakage criteria.

Note 10. Repair Criteria
SR 3.4.18.2 of the proposed license amendments requires that tubes that exceed approved tube repair criteria be repaired in accordance with approved methods. SG tubes experiencing a damage form or mechanism for which no depth sizing capability exists are "repaired/plugged on detection" and their integrity should be assessed.

There are presently no NRC approved alternate repair criteria listed in the Catawba TS. At Catawba, tubes with imperfections extending > 40% through wall must be plugged. No changes to the applicable tube plugging criterion are being proposed in conjunction with the proposed amendments.

Note 11. Exceeding Performance or Repair Criteria
The RCS Operational Leakage and Steam Generator Tube

Integrity TS require the licensee to monitor SG performance

against performance criteria in accordance with the SGP.

During plant operation, monitoring is performed using the operational leakage criterion. Exceeding this criterion will lead to a plant shutdown in accordance with TS 3.4.13. Once shut down, the SGP will ensure that the cause of the operational leakage is determined and corrective actions to prevent recurrence are taken. Operation may resume when the requirements of the SGP have been met. This requirement is unchanged from the existing TS.

Also, during plant operation the licensee may discover an error or omission that indicates a failure to implement a required plugging or repair during a previous SG inspection. Under these circumstances, the licensee is expected to take the actions required by Condition A in the Steam Generator Tube Integrity TS. If a performance criterion has been exceeded, a principal safety barrier has been challenged and 10 CFR 50.72(b) (3) (ii) (A) and 10 CFR 50.73(a) (2) (ii) (A) require NRC notification and the submittal of a report containing the cause and corrective actions to prevent recurrence. The SGP additionally requires that the report contain information on the performance criteria exceeded and the basis for the planned operating cycle. Irhe existing TS only address operational leakage during operations and therefore do not include the proposed requirement.

During shutdown periods, the operational leakage criterion is not applicable, and the SGs will be inspected as required by SR 3.4.18.1. A condition monitoring assessment of the Iras found" condition of the SG tubes will be performed to determine the condition of the SGs with respect to the structural integrity and accident induced leakage performance criteria. If the performance criteria are not met, the SGP requires ascertaining the cause and determining corrective actions to prevent recurrence.

Attachment 3 Page 19

Operation may resume when the requirements of the SGP have been met.

The proposed change to the TS actions required upon exceeding the operational leakage criterion is conservative with respect to the existing TS as explained in Note 2 above.

The existing TS do not address actions required while operating if it is discovered that the structural integrity or the accident induced leakage performance criteria or a repair criterion are exceeded, so the proposed change is conservative with respect to the existing TS.

If performance or repair criteria are exceeded while shutdown, required actions consist of repairing or plugging the affected tubes. If the number of degraded tubes exceeds 1% of those inspected in any SG, a report will be submitted to the NRC per TS 5.6.8. The changes in the required reports are discussed in Note 13 below.

Note 12. Repair Methods
Proposed SR 3.4.18.2 requires that tubes that satisfy approved tube repair criteria be plugged or repaired in accordance with the SGP. At Catawba, tubes with imperfections extending > 40% through wall must be plugged.

SG tubes experiencing a damage form or mechanism for which no depth sizing capability exists are "repaired/plugged on detection" and their integrity is assessed. This requirement is unchanged by the proposed amendments.

The means of obtaining NRC approval of new repair methods is changed by these amendments. The NRC staff currently approves repair methods on a plant by plant basis. The staff reviews each plant specific license amendment request and approves the proposal on a plant specific basis, including a plant specific TS change that, as a minimum, lists each method and may include specific technique requirements.

The proposed amendments do not always require a license amendment for adoption of new repair methods. In addition to license amendment approvals, repair methods that are specifically listed in editions or addenda of ASME Section XI, IWA-4720, approved by 10 CFR 50.55a, may also be used if the limiting design parameters identified for the repair

method envelop Catawba's design. Therefore, this proposed approach still requires NRC approval of SG repair methods, but allows that approval to come by way of either an ASME Code change or a TS amendment.

These proposed amendments also affect how repair methods are listed in the TS. It is proposed that repair methods approved by license amendment be listed in Administrative TS 5.5.9, while repair methods approved through ASME Code endorsement do not need to be listed in the TS. Instead, Code authorized repair methods will be included in the Bases for the Steam Generator Tube Integrity TS in order to document that they have been evaluated for use at Catawba.

Note that SG plug designs do not require NRC review and therefore plugging is not considered a repair in the context of this requirement.

The proposed approach is a change to the TS, but is not a significant relaxation of the requirements. NRC approval of all repair methods will still be required because NRC endorsement of the applicable ASME Code is required by 10 CFR 50.55a before any repair methods listed in the Code can be used. In addition, these amendment requests remove unnecessary regulatory burden by allowing generic approval of repair methods that are applicable at more than one plant.

Note 13. Reporting Requirements
The current TS require the following reports:

- A report listing the number of tubes plugged in each SG submitted within 15 days of the end of the inspection

- A SG inspection results report submitted within 12 months after the inspection

- A report describing the results of inspections of SG tubes which fall into category C-3 submitted within 30 days prior to the restart of the unit following the inspection

The proposed amendments to TS 5.6.8 replace the 15-day and the SG inspection reports with one report required within 120 days if greater than 1% of the tubes inspected in any one SG exceed a repair criterion. The proposed report also contains more information than the old SG inspection report. This provision limits the reports submitted to the NRC to those documenting more extensive degradation and

requires that the reports that are submitted provide more substantive information and be sent earlier (120 days versus 12 months). This allows the NRC to focus its attention on the more significant conditions earlier and with more data and analyses.

If a performance criterion has been exceeded a principal safety barrier has been challenged and 10 CFR 50.72(b)(3)(ii)(A) and 10 CFR 50.73(a)(2)(ii)(A) require NRC notification and the submittal of a report containing the cause and corrective actions to prevent recurrence. The 10 CFR requirements are therefore unaffected.

The proposed reporting requirements are an improvement as compared to those required by the existing TS. The proposed reporting requirements are more useful in identifying the degradation mechanisms and in determining their effects. In the unlikely event that a performance criterion is not met, NEI 97-06 requires submitting additional information on the root cause of the condition and the basis for the next operating cycle.

Consistent with the proposed amendments, the changes to the reporting requirements are performance based. The new requirements remove the burden of unnecessary reports from both the NRC and the licensee, while ensuring that critical information related to problems and significant tube degradation is reported more completely and, when required, more expeditiously than under the current TS.

Note 14. Definitions
The proposed SG Tube Integrity TS Bases use a number of terms that are important to the function of a SGP. These terms are not in the existing TS and are not proposed for inclusion in the amended ~S, but are captured in the proposed Bases for the SG Tube Integrity TS 3.4.18. As part of the Bases for TS 3.4.18, they will be controlled by the TS Bases Control Program under the provisions of 10 CFR 50.59.

The terms are defined and explained below.

1) Accident induced leakage rate means the primary to secondary leakage rate occurring during postulated accidents other than a SG tube rupture. This includes the primary to secondary leakage rate existing immediately prior to the accident plus additional

priniary to secondary leakage induced during the accident.

Primary to secondary leakage is a factor in the dose releases outside containment resulting from a limiting design basis accident. The potential primary to secondary leak rate during postulated design basis accidents must not cause radiological dose consequences in excess of the 10 CFR 100 guidelines for offsite doses, or the GDC 19 requirements for control room personnel, or other NRC approved licensing basis (e.g., 10 CFR 50.67).

2) Burst is defined as the gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation.

Since a burst definition is a required component for condition monitoring, a definition that can be analytically defined and is capable of being assessed via in situ and laboratory testing is required. Furthermore, the definition must be consistent with ASME Code requirements, and apply to most forms of tube degradation.

The definition developed for tube burst demonstrates accord with the testimony of James Knight [8], and compliance with the historical guidance of draft Regulatory Guide 1.121 [9]. The definition of burst per these documents is in relation to gross failure of the pressure boundary (e.g., 11the degree of loading required to burst or collapse a tube wall is consistent with the design margins in Section III of the ASME B&PV Code [l0]~). Burst, or gross failure, according to the Code would be interpreted as a catastrophic failure of the pressure boundary.

The above definition of burst was chosen for a number of reasons:

- The burst definition supports field application of the condition monitoring process. For example, verification of structural integrity during condition monitoring may be accomplished

via in situ testing. Since these tests do not have the capability to provide an unlimited water supply, or the capability to maintain pressure under certain leakage scenarios, opening area may be more a function of fluid reservoir rather than tube strength. Additionally, in situ designs with bladders may not be reinforced. In certain cases, the bladder may rupture when tearing or extension of the defect has not occurred. This condition may simply mean the opening of the flanks of the defect was sufficient to permit extrusion of the bladder, and that the actual, or true, burst pressure was not achieved during the test. The burst definition addresses this issue.

The definition does not characterize local instability or "ligament pop-through11, as a burst. The onset of ligament tearing need not coincide with the onset of a full burst. For example, an axial crack about 0.5" long with a uniform depth at 98% of the tube wall would be expected to fail the remaining ligament (i.e., extend the crack tip in the radial direction) due to deformation during pressurization at a pressure below that required to cause extension at the tips in the axial direction. Thus, this would represent a leakage situation as opposed to a burst situation and a factor of safety of 3.0 against crack extension in the axial direction may still be demonstrated. Similar conditions have been observed for deep wear indications.

3) Normal steady state full power operation is defined as the conditions existing during Mode 1 operation at the maximum steady state reactor power as defined in the design or eouipment specification. Changes in design parameters such as plugging or sleeving levels, primary or secondary modifications, or T~0t should be assessed and their effects on differential pressure should be included if significant.

The definition of normal full power operation is important as it relates to application of the safety factor of 3.0 in the structural integrity performance criterion. The criterion requires .1... retaining a safety factor of 3.0 under normal steady state full power operation primary to secondary pressure

differential ...". The application of the safety factor of 3.0 to normal steady state full power operation is founded on past NRC positions, accepted industry practice, and the intent of the ASME Code for original design and evaluation of inservice components. The assumption of normal steady state full power operating pressure differential has been consistently used in the analysis, testing and verification of tubes with stress corrosion cracking for verifying a safety factor of 3.0 against burst. Additionally, the 3 times differential pressure criterion is measurable through the condition monitoring process.

The actual operational parameters may differ between cycles. As a result of changes to these parameters, reaching the differential pressure in the equipment specification may not be possible during plant operations. Evaluating to the pressure in the design or equipment specification in these cases would be an unnecessary conservatism. Therefore, the definition allows adjustment of the 3 times differential pressure limit for changes in these parameters when necessary. Further guidance on this adjustment is provided in Appendix M of the EPRI Steam Generator Integrity Assessment Guideline [2].

4) Repair cri~eria are those NDE measured parameters at or beyond which the tube must be repaired or removed from service by plugging.

Thbe repair criteria are established for each active degradation mechanism. Tube repair criteria are either the standard through wall (TW) depth based criterion (e.g., 40% Tw for most plants) or TW depth based criteria for repair techniques approved by the NRC, or other alternate repair criteria (ARC) approved by the NRC such as a voltage based repair limit per Generic Letter 95-05 [12]. A SG degradation specific management (SGDSM) strategy is followed to develop and implement an ARC.

Tubes identified with a damage form or mechanism for which no depth sizing capability exists, are 11repaired/plugged on detection" and their integrity is assessed.

Attachment 3 Page 25

An ARC methodology will be reviewed and approved by the NRC prior to its first time use at a licensed facility. Subsequent use of a generically approved ARC at Catawba will be justified by an evaluation that shows that Catawba's design falls within the parameters defined by the NRC in the SER approving the ARC. There are presently no ARCs approved for use at Catawba.

5) Repair methods are those means used to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. Plugging a SG tube is not a repair.

The purpose of a repair is typically to reestablish or replace the reactor coolant pressure boundary. Repair methods are qualified and implemented in accordance with industry standards. The qualification of the repair techniques considers the specific SG conditions and mockup testing.

New repair methods will be reviewed and approved by the NRC prior to use at a licensed facility. Note that in the context of the TS, "plug on detection" is not considered a repair.

6) SG tubing refers to the entire length of the tube, including the tube wall and any repairs to it, between the tube to tubesheet weld at the tube inlet and the tube to tubesheet weld at the tube outlet. The tube to tubesheet weld is not considered part of the tube.

This definition ensures that all portions of SG tubes that are part of the RCS pressure boundary, with the exception of the tube to tubesheet weld, are subject to SGP requirements. The definition is also intended to exclude tube ends that cannot be NDE inspected by eddy current. If there are concerns in the area of the tube end, they will be addressed by NDE techniques if possible or by using other methods if necessary.

For the purposes of SG tube integrity inspection, any weld metal in the area of the tube end is not considered part of the tube. This is necessary since the acceptance requirements for tubing and weld metals are different.

The above terms are not included in either the existing or proposed 'I'S, but since they are included in the proposed Bases whose changes are controlled by the TS Bases Control Program under the provisions of 10 CFR 50.59, the proposed change is considered an improvement over the existing TS requirements.

E. Safety Analysis

The proposed amendments do not affect the design of the SGs, their method of operation, or primary coolant chemistry controls. The proposed changes are an improvement to the existing SG inspection requirements and provide additional assurance that the plant licensing basis will be maintained between SG inspections. The proposed changes do not adversely impact any previously evaluated design basis accident.

A SG tube rupture event is one of the design basis accidents that are analyzed as part of a plant's licensing basis. In the analysis of a SG tube rupture event, a bounding primary to secondary leakage rate equal to the operational leakage rate limits in the licensing basis plus the leakage rate associated with a double ended rupture of a single tube is assumed.

For design basis accidents such as main steam line break, rod ejection accident, reactor coolant pump locked rotor accident, and uncontrolled rod withdrawal accident, the tubes are assumed to retain their structural integrity (i.e., they are assumed not to rupture). These analyses assume that primary to secondary leakage through each SG is 150 gallons per day. For accidents that do not involve fuel damage, the reactor coolant activity levels are at the TS values. For accidents that do involve fuel damage, the primary coolant activity values are a function of the amount of activity released from the damaged fuel.

The consequences of these design basis accidents are, in part, functions of the radioactivity levels in the primary coolant and the accident primary to secondary leakage rates. As a result, limits are included in the plant TS for operational leakage and for dose equivalent 1131 in primary coolant to ensure the plant is operated within its analyzed condition.

The primary coolant activity limit and its assumptions are not affected by the proposed TS changes.

The TS changes proposed by these amendments are in general a significant improvemen~ over existing requirements. They replace an outdated prescriptive TS with one that references SGP requirements that incorporate the latest knowledge of SG tube degradation morphologies and the techniques developed to manage them.

The requirements proposed in these amendments are more effective in detecting SG degradation and prescribing corrective actions than are the existing TS. As a result, these proposed changes will result in added assurance of the function and integrity of SG tubes.

Therefore, the proposed changes do not affect the consequences of a SG tube rupture or any other design basis accident and the likelihood of a SG tube rupture is reduced.

F. Conclusions

The proposed license amendments will provide greater assurance of SG tube integrity than that offered by the current TS. The proposed requirements are performance based and provide the flexibility to adopt new technology as it matures. These changes are consistent with the guidance in NEI 97-06, Steam Generator Program Guidelines.

Adopting the changes proposed by these license amendments will provide added assurance that SG tubing will remain capable of fulfilling its specific safety function of maintaining RCPB integrity.

G. References

1. PWR Steam Generator Examination Guidelines, EPRI

Report TR-107569.

2. Steam Generator Integrity Assessment Guideline, EPRI Report TR-l0762l.

3. Steam Generator In Situ Pressure Test Guidelines, EPRI Report TR-107620.

4. PWR Primary-to-Secondary Leak Guidelines, EPRI Report IIIR-104788.

5. PWR Primary Water Chemistry Guidelines, EPRI Report TR-105714.

6. PWR Secondary Water Chemistry Guidelines, EPRI Report TR-102134.

7. Steam Generator Program Guidelines, NEI 97-06.

8. Testimony of James Knight before the Atomic Safety and Licensing Board, Docket Nos. 50-282 and 50-306, January 1975.

9. Draft Regulatory Guide 1.121, Basis for Plugging Degraded Steam Generator Tubes, August 1976.

10. ASME B&PV Code, Section III, Rules for Construction of Nuclear Facility Components.

11. Experience of US and Foreign PWR Steam Generators with Alloy 6OOTT and Alloy 690TT Tubes and Sleeves, EPRI Report R-5515-00-2, June 5, 2002.

12. Generic Letter 95-05, voltage Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking, August 3, 1995.

ATTACHNENT 4

NO SIGNIFICANT HAZARDS CONSIDERATIONS ANALYSIS

The proposed changes have been evaluated against the standards in 10 CFR 50.92 and have been determined to not involve a significant hazards consideration. The proposed amendments require a SG Program that defines a performance based approach to maintaining SG tube integrity. The SG Program includes performance criteria that define the basis for SG tube integrity and provide reasonable assurance that the SG tubing will remain capable of fulfilling its safety function of maintaining RCPB integrity. The SG Program is an improvement over the existing requirements. The proposed amendments add a new TS for SG Tube Integrity (3.4.18), and revise the TS for RCS Operational Leakage (3.4.13), SG Tube Surveillance Program (5.5.9), and SG Thbe Inspection Report (5.6.8).

Operation of the facility in accordance with the proposed amendments:

1. Would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes require a SG Program that includes performance criteria that will provide reasonable assurance that the SG tubing will retain integrity over the full range of operating conditions (including startup, operation in the power range, hot standby, cooldown, and all anticipated transients included in the design specification) and design basis accidents.

The SG performance criteria are based on tube structural integrity, accident induced leakage, and operational leakage.

The structural integrity performance criterion is a new requirement. It is included in the proposed SG Program administrative TS 5.5.9.

The accident induced leakage criterion is a new requirement. It is included in the proposed SG Program administrative TS 5.5.9.

The operational leakage criterion is equivalent to the existing requirement. Its limit is part of the proposed RCS Operational Leakage TS 3.4.13.

Attachment 4 Page 1

A SG tube rupture event is one of the design basis accidents analyzed as part of Catawba's licensing basis. In the analysis of a SG tube rupture event, a bounding primary to secondary leakage rate equal to the operational leakage rate limit in the licensing basis plus the leakage rate associated with a double-ended rupture of a single tube is assumed. For other design basis accidents, the tubes are assumed to retain their structural integrity (i.e., they are assumed not to rupture). These analyses assume that primary to secondary leakage through each SG is 150 gallons per day.

The accident induced leakage criterion introduced by the proposed changes accounts for tubes that may leak during design basis accidents. The accident induced leakage criterion limits this leakage to no more than the value assumed in the accident analysis. The SG performance criteria proposed as part of these TS amendments identify the standards against which tube integrity is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining RCPB integrity throughout each operating cycle and in the unlikely event of a design basis accident. The performance criteria are only a part of the SG Program required by the proposed changes to TS 5.5.9. The program, defined by NEI 97-06, "Steam Generator Program Guidelines," includes a framework that incorporates a balance of prevention, inspection, evaluation, repair, and leakage monitoring.

Probability of an Accident

The TS proposed by these license amendments define the actions required upon failure to maintain SG tube integrity and the surveillances necessary to verify that tube integrity is maintained. The proposed administrative TS contain performance criteria, repair criteria, repair methods, maximum SG inspection intervals, and reporting requirements. The set of TS proposed is a significant improvement over the existing SG TS.

In addition, the SG Program required by these amendments includes provisions important in satisfying the TS requirements. The topics addressed by the SG Program include:

- SG performance criteria, including an operational leakage limit,

- SG repair criteria and repair methods,

- SG inspection intervals, and

- Performance based SG inspections that include pre-inspection degradation assessments, condition monitoring assessments, operational assessments, and non-destructive examination technique requirements.

These SG Program provisions establish requirements that are an improvement as compared to the requirements in the existing TS. As an example, the SG Program requires an operational assessment that defines the maximum SG inspection interval that provides reasonable assurance that the performance criteria will continue to be met at the next inspection. The actual inspection interval is always chosen to be less than the interval determined by the operational assessment. The existing TS have no similar requirement. As a result, the function and integrity of the tubes are maintained with greater assurance and the probability of a SG tube rupture is decreased.

Consequences of an Accident

The consequences of design basis accidents are, in part, functions of the dose equivalent 1131 in the primary coolant and the primary to secondary leakage rates resulting from an accident. Therefore, limits are included in the plant TS for operational leakage and for dose equivalent 1131 in primary coolant to ensure the plant is operated within its analyzed condition.

The analysis of the associated design basis accidents assumes that the initial primary to secondary leak rate is 150 gallons per day in each SG (except for the ruptured SG in a SG tube rupture), and that the reactor coolant activity levels of dose equivalent 1131 are at the TS values before the accident. The TS limits, license conditions, and other controls on 1131 are unchanged by these amendment requests. These other controls include License Amendments 159 and 151 for Catawba Units 1 and 2, respectively, and the Catawba license amendment request submittal dated May 9, 2002, which is presently being reviewed by the NRC.

In addition, the proposed amendments include a new performance criterion for accident induced leakage that requires that the primary to secondary leakage resulting

from an accident other than a SG tube rupture not exceed the value assumed in the dose analyses (150 gallons per day through each SG).

Since the proposed operational leakage limit is equivalent to the existing value, and since the proposed amendments include a new performance criterion for accident induced leakage, the proposed amendments will not increase the consequences of an accident.

From the above discussion, it is concluded that the proposed amendments do not affect the design of the SGs, their method of operation, or primary coolant chemistry controls. The proposed approach updates the existing TS and enhances the requirements for SG inspections. The proposed TS changes do not adversely impact any other previously evaluated design basis accident and represent an improvement over the existing TS. Therefore, the proposed changes do not affect the consequences of a SG tube rupture accident and the probability of such an accident is reduced. In addition, the proposed changes do not affect the consequences of other accidents.

2. Would not create the possibility of a new or different kind of accident from any other accident previously evaluated.

The proposed performance based requirements are an improvement over the requirements imposed by the existing TS. Implementation of the proposed SG Program will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. The result of the implementation of the SG Program will be an enhancement of SG tube performance. Primary to secondary leakage that may be experienced during all plant conditions will be monitored to ensure it remains within current accident analysis assumptions.

The proposed amendments do not affect the design of the SGs, their method of operation, or primary or secondary coolant chemistry controls. In addition, the proposed changes do not impact any other plant system or component. The changes enhance SG inspection requirements. Therefore, the proposed changes do not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Would not involve a significant reduction in a margin of safety.

The SG tubes in pressurized water reactors are an integral part of the RCPB and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the RCPB, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes also isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of a SG is maintained by ensuring the integrity of its tubes. SG tube integrity is a function of the design, environment, and physical condition of the tube. The proposed license amendments do not affect tube design or operating environment. The proposed changes are expected to result in an improvement in the tube integrity by implementing the SG Program to manage SG tube inspection, assessment, repair, and plugging. The requirements established by the SG Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the existing TS.

For the above reasons, the margin of safety is not changed and overall plant safety will be enhanced by the proposed revisions to the TS.

Attachment 4 Page 5

ATTACHMENT 5

ENVIRONMENTAL ANALYSIS

Pursuant to 10 CFR 51.22(b), an evaluation of these license amendment requests has been performed to determine whether or not they meet the criteria for categorical exclusion set forth in 10 CFR 51.22(c) (9) of the regulations.

Implementation of these amendments will have no adverse impact upon the Catawba units; neither will they contribute to any additional quantity or type of effluent being available for adverse environmental impact or personnel exposure.

It has been determined there is:

1. No significant hazards consideration,

2. No significant change in the types, or significant increase in the amounts, of any effluents that may be released of fsite, and

3. No significant increase in individual or cumulative occupational radiation exposures involved.

Therefore, these amendments to the Catawba TS and Bases meet the criteria of 10 CFR 51.22(c) (9) for categorical exclusion from an environmental impact statement.

Attachment 5 Page 1